Office for
Nuclear Regulation

New Reactors Division – Generic Design Assessment

Step 4 Assessment of Chemistry for the UK HPR1000 Reactor

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EXECUTIVE SUMMARY

This report presents the findings of my assessment of the Chemistry aspects of the UK HPR1000 reactor design undertaken as part of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA). My assessment was carried out using the Pre-Construction Safety Report (PCSR) and supporting documentation submitted by the Requesting Party (RP).

The objective of my assessment was to make a judgement, from a Chemistry perspective, on whether the generic UK HPR1000 design could be built and operated in Great Britain, in a way that is acceptably safe and secure (subject to site specific assessment and licensing), as an input into ONR's overall decision on whether to grant a Design Acceptance Confirmation (DAC).

The scope of my GDA assessment was to review the safety aspects of the generic UK HPR1000 design by examining the claims, arguments and supporting evidence in the safety case. My GDA Step 4 assessment built upon the work undertaken in GDA Steps 2 and 3, and enabled a judgement to be made on the adequacy of the chemistry information contained within the PCSR and supporting documentation.

My assessment focussed on a number of main themes of relevance to chemistry control. These were control of coolant reactivity, protection of the structural materials, maintaining fuel integrity and performance, minimisation of out of core radiation fields and minimisation of releases during fault/accident conditions. The aspects of the generic UK HPR1000 safety case that I focussed my assessment on were set out in my assessment plan produced prior to the start of Step 4.

The main conclusions from my assessment are:

- In general, the scope, structure and content of the generic safety case meet my expectations for this stage of the project. From a Chemistry perspective, an adequate PCSR has been produced, which provides a summary of and links to the underlying evidence. However, further work will need to be undertaken by the licensee to develop the chemistry aspects of the safety case.
- The RP has identified a suitable set of claims on the operating chemistry for all modes of operation. In the majority of cases, the RP has provided adequate supporting evidence to demonstrate that the claims can be achieved by the generic design.
- The RP has provided an appropriate demonstration that the generic plant design and engineering is adequate to achieve effective control of chemistry for relevant systems (including dosing, monitoring and clean-up), and to maintain the chemistry within the limits defined within the safety case.
- The different chemistry requirements likely to be necessary during different operating modes, and during different stages of the plant's lifetime have been suitably considered. The major chemistry parameters, which would be expected to form part of the plant operating rules, have been identified and limits have been appropriately justified.
- The RP has appropriately considered the through life performance of the chemistry related Structures Systems and Components (SSCs), and the effects of the chemistry regime on the susceptibility to material degradation mechanisms, in making suitably justified materials selection decisions.
- The generation, transport and accumulation of radioactivity has been suitably analysed and quantified and the safety case successfully developed to provide an adequate demonstration that radioactivity will be reduced So Far As Is Reasonably Practicable (SFAIRP).

- An adequate demonstration has been made that the effects of chemistry during fault and accident conditions, including the generation, transport and behaviour of radionuclides, reactive species and flammable gases, are understood and that risks will be reduced SFAIRP.
- For those risks on which chemistry can have an influence, an appropriate overall demonstration has been made that chemistry effects are understood and that risks will be reduced SFAIRP.

These conclusions are 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 of the hierarchy of the generic UK HPR1000 safety case documentation.
- Independent information, reviews and analysis of key aspects of the generic safety case undertaken by Technical Support Contractors (TSCs).
- Detailed technical interactions on many occasions with the RP, alongside the assessment of the responses to Regulatory Queries (RQs) and Regulatory Observations (ROs) raised during the GDA.

A number of matters remain, which I judge are appropriate for the licensee to consider and take forward in its site-specific safety submissions. These matters do not undermine the generic UK HPR1000 design and safety submissions, but are primarily concerned with the provision of site-specific safety case evidence which will become available as the project progresses through the detailed design, construction and commissioning stages. These matters have been captured in 21 Assessment Findings.

Overall, based on my assessment undertaken in accordance with ONR’s procedures, the claims, arguments and evidence, laid down within the PCSR and supporting documentation submitted as part of the GDA process, present an adequate safety case for the generic UK HPR1000 design. I recommend that from a Chemistry perspective a DAC may be granted.

LIST OF ABBREVIATIONS

AAD [SSFS]	Start-up and Shutdown Feedwater System
ABP [LPFHS]	Low Pressure Feedwater Heater System
ACO [FHDRS]	Feedwater Heater Drain Recovery System
ADG [FDTGSS]	Feedwater Deaerating Tank and Gas Stripper System
AFCEN	French Association for Design, Construction and Surveillance Rules of Nuclear Power Plants Components
AHP [HPFHS]	High Pressure Feedwater Heater System
AICC	Adiabatic Isochoric Complete Combustion
ALARP	As Low As Reasonably Practicable
APG [SGBS]	Steam Generator Blowdown System
ARE [MFFCS]	Main Feedwater Flow Control System
ASG [EFWS]	Emergency Feedwater System
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATE [CPS]	Condensate Polishing System
AVT	All Volatile Treatment
BAST	Boric Acid Storage Tank
BMS	Business Management System
BOC	Beginning of Cycle
BFX	Fuel Building
BNX	Nuclear Auxiliary Building
BRX	Reactor Building
BSA	Safeguard Building A
BWX	Radioactive Waste Processing Building
CEX [CES]	Condensate Extraction System
CFD	Computational Fluid Dynamics
CFT	Cold Functional Testing
CGN	China General Nuclear Power Corporation
CILC	Crud Induced Localised Corrosion
CIPS	Crud Induced Power Shift
CPs	Corrosion Products
CPR	Chinese PWR
CRDM	Control Rod Drive Mechanism
CRF [CWS]	Circulating Water System
DAC	Design Acceptance Confirmation
DBC	Design Basis Condition
DEC	Design Extension Condition
DEI	Dose Equivalent Iodine

DSR	Design Substantiation Report
DZA	Depleted Zinc Acetate
EBA	Enriched Boric Acid
EHR [CHRS]	Containment Heat Removal System
EPRI	Electric Power Research Institute (US)
EUF [CFES]	Containment Filtration and Exhaust System
EUH [CCGCS]	Containment Combustible Gas Control System
FAC	Flow Accelerated Corrosion
FCG3	Fangchenggang Nuclear Power Plant Unit 3
GCT [TBS]	Turbine Bypass System
GDA	Generic Design Assessment
GNSL	General Nuclear System Ltd.
H-AVT	High pH All Volatile Treatment
HFT	Hot Functional Testing
HIC	High Integrity Component
IAEA	International Atomic Energy Agency
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IB-LOCA	Intermediate Break Loss of Coolant Accident
IGA	Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
IPC	Iodine Partition Coefficient
IRWST	In-containment Refuelling Water Storage Tank
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
ISI	In-Service Inspection
IVR	In-Vessel Retention
KRT [PRMS]	Plant Radiation Monitoring System
LAS	Low Alloy Steel
LB-LOCA	Large Break Loss of Coolant Accident
LCO	Limits and Conditions of Operation
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Offsite Power
LWR	Light Water Reactor
MCL	Main Coolant Line
MCCI	Molten Core Concrete Interaction
MDSL	Master Document Submission List
MSL	Main Steam Line
MSR	Moisture Separator Reheater
MSS	Main Steam System
NEA	Nuclear Energy Agency (within OECD)

NI	Nuclear Island
ODSCC	Outer Diameter Stress Corrosion Cracking
OECD	Organisation for Economic Cooperation and Development
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
ORE	Operator Radiation Exposure
PAR	Passive Autocatalytic Recombiner
PCER	Pre-construction Environmental Report
PCSR	Pre-construction Safety Report
PMC [FHSS]	Fuel Handling and Storage System
PTR [FPCTS]	Fuel Pool Cooling and Treatment System
PWR	Pressurised Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
PZR	Pressuriser
RBS [EBS]	Emergency Boration System
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCP [RCS]	Reactor Coolant System
RCV [CVCS]	Chemical Volume and Control System
REA [RBWMS]	Reactor Boron Water Make-up System
REN [NSS]	Nuclear Sampling System
RG	Regulatory Guide
RGP	Relevant Good Practice
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RIS [SIS]	Safety Injection System
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPE [VDS]	Vent and Drain System
RPFT	Reactor Pit Flooding Tank
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
RRI [CCWS]	Component Cooling Water Systems
RVI	Reactor Vessel Internals
SAP(s)	Safety Assessment Principle(s)
SB-LOCA	Small Break LOCA
SCC	Stress Corrosion Cracking
SDM	System Design Manual

SEC [ESWS]	Essential Service Water System
SED [DWDS (NI)]	Dematerialised Water Distribution System (Nuclear Island)
SER [DWDS (CI)]	Demineralised Water Distribution System (Conventional Island)
SFAIRP	So Far As Is Reasonably Practicable
SFIS	Spent Fuel Interim Storage
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIH [CRDS (NI)]	Chemical Reagents Distribution System Nuclear Island
SIR [CRIS]	Chemical Reagents Injection System
SIT [FCSS]	Feedwater Chemical Sampling System
SNS	Secondary Neutron Source
SRE [SRS]	Sewage Recovery System
SSCs	Structures, Systems and Components
TAG	Technical Assessment Guide(s)
TBS [GCT]	Turbine Bypass System
TEG [GWTS]	Gaseous Waste Treatment System
TEP [CSTS]	Coolant Storage and Treatment System
TER [NLWDS]	Nuclear Island Liquid Waste Discharge System
TES [SWTS]	Solid Waste Treatment System
TEU [LWTS]	Liquid Waste Treatment System
TMS	Tritium Management Strategy
TSC	Technical Support Contractor
TSP	Trisodium Phosphate
TT	Thermally Treated alloy (specifically Alloy 690)
WENRA	Western European Nuclear Regulators' Association
VCT	Volume Control Tank
VGB	Verenigate Grosskraftwerke Betreiber (Federation of Large Power Station Operators, Germany)
VPU [MSDS]	Main Steam and Drainage System for CI
VVP [MSS]	Main Steam System
V&V	Verification and Validation

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1 INTRODUCTION

1.1 Background

1. This report presents my assessment conducted as part of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA) for the generic UK HPR1000 design within the topic of Chemistry.

2. The UK HPR1000 is a Pressurised Water Reactor (PWR) design proposed for deployment in the United Kingdom (UK). General Nuclear System Ltd is a UK-registered company that was established to implement the GDA on the UK HPR1000 design on behalf of three joint Requesting Parties (RP), i.e. China General Nuclear Power Corporation (CGN), EDF SA and General Nuclear International Ltd.

3. GDA is a process undertaken jointly by the ONR and the Environment Agency. Information on the GDA process is provided in a series of documents published on the joint regulators' website (www.onr.org.uk/new-reactors/index.htm). The outcome from the GDA process sought by the RP is a Design Acceptance Confirmation (DAC) from ONR and a Statement of Design Acceptability from the Environment Agency.

4. The GDA for the generic UK HPR1000 design followed a step-wise approach in a claims-argument-evidence hierarchy which commenced in 2017. Major technical interactions started in Step 2 of GDA which focussed on an examination of the main claims made by the RP for the UK HPR1000. In Step 3 of GDA, the arguments which underpin those claims were examined. The Step 2 reports for individual technical areas, and the summary reports for Steps 2 and 3 are published on the joint regulators' website. The objective of Step 4 of GDA was to complete an in-depth assessment of the evidence presented by the RP to support and form the basis of the safety and security cases.

5. The full range of items that form part of my 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) and whether the proposed design ensures risks are reduced So Far As Is Reasonably Practicable (SFAIRP) (Ref. 2).
   - Reviewing details of the RP's design controls and quality control arrangements to secure compliance with the design intent.
   - Establishing whether the system performance, safety classification, and reliability requirements are substantiated by a more detailed engineering design.
   - Assessing arrangements for ensuring and assuring that safety claims and assumptions will be realised in the final as-built design.
   - Resolution of identified nuclear safety and security issues, or identifying paths for resolution.

6. The purpose of this report is therefore to summarise my assessment in the Chemistry topic which provides an input to the ONR decision on whether to grant a DAC, or otherwise. This assessment was focused on the submissions made by the RP throughout GDA, including those provided in response to the Regulatory Queries (RQs) and Regulatory Observations (ROs) (Ref. 3, Ref. 4). The ROs issued to the RP are published on the GDA's joint regulators' website, together with the corresponding resolution plans.

1.2 Scope of this Report

7. This report presents the findings of my assessment of the chemistry aspects of the generic UK HPR1000 design, undertaken as part of GDA. I carried out my assessment using the Pre-construction Safety Report (PCSR) and supporting documentation submitted by the RP (Ref. 5). My assessment was focussed on considering whether the generic safety case provides an adequate justification for the generic UK HPR1000 design, in line with the objectives for GDA.

1.3 Methodology

8. The methodology for my assessment follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (Ref. 6).

9. My assessment was undertaken in accordance with the requirements of ONR‘s How2 Business Management System (BMS). ONR's SAPs, together with supporting Technical Assessment Guides (TAG) were used as the basis for my assessment (Ref. 2, Ref. 6). Further details are provided in Section 2. The outputs from my assessment are consistent with ONR's GDA guidance to RPs (Ref. 1).

2 ASSESSMENT STRATEGY

10. The strategy for my assessment of the Chemistry aspects of the generic UK HPR1000 design and safety case is set out in this section. This identifies the scope of the assessment and the standards and criteria that have been applied.

2.1 Assessment Scope

11. A detailed description of my approach to this assessment can be found in assessment plan ONR-GDA-UKHPR1000-AP-19-003 (Ref. 7).

12. I considered all of the main submissions within the remit of my assessment scope, to various degrees of breadth and depth. I chose to concentrate my assessment on those aspects that I judged to have the greatest safety significance, or where the hazards appeared least well controlled. My assessment was also influenced by the claims made by the RP, my previous experience of similar systems for reactors and other nuclear facilities, and any identified gaps in the original submissions made by the RP. A particular focus of my assessment has been the RQs and ROs raised as a result of my ongoing assessment, and the resolution thereof.

2.2 Sampling Strategy

13. In line with ONR's guidance I chose a sample of the RP's submissions to undertake my assessment (Ref. 6). In doing so, I focussed on matters which I judged to be the most safety significant, where significant design or safety case changes may have been needed, or where there was a potential for a significant matter to be revealed that may prevent ONR issuing a DAC.

14. Consistent with my assessment plan, my assessment focussed on whether the generic safety case provides an adequate justification for the generic design of UK HPR1000, for those aspects relating to Chemistry (Ref. 7). For the purposes of this assessment, the definition of Chemistry is considered 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.

15. The main themes I considered throughout my assessment were therefore control of coolant reactivity, protection of the structural materials, maintaining fuel integrity and performance, minimisation of out of core radiation fields and minimisation of releases during fault/accident conditions. The key topic areas that I assessed are set out in my assessment plan (Ref. 7). I structured my assessment by system, focussing on chemistry control in the primary circuit, secondary circuit and key auxiliary systems including the Spent Fuel Pool (SFP) and the radioactive waste processing systems. My approach to accident chemistry assessment was to sample accident scenarios where chemistry was a key factor, either as part of the accident source term, or where it was identified as part of accident mitigation.

16. I also followed up on a number of key items identified in my Step 3 assessment, including the minimisation of risks relating to fuel deposits, radioactivity minimisation and the control of boron chemistry (Ref. 8). Each assessment topic is described in detail in Section 4 of this report.

2.3 Out of Scope Items

17. The following items were outside the scope of my assessment.

- Site-specific and/or detailed design aspects, which include detailed methodologies for commissioning and the detailed design of the zinc injection system.
- Site-specific operational matters, such as detailed chemical specifications. For example, during GDA I have judged the adequacy of the RP's approach to deriving and justifying the chemistry-related limits and conditions necessary in the interests of safety, in terms of the suitability of the list of chemical parameters and the bounding limits proposed. However, I have not considered any details relating to the way in which the licensee may develop the chemistry-related operating rules, or incorporate other limits such as action levels.

2.4 Standards and Criteria

18. The relevant standards and criteria adopted within this assessment are principally the SAPs, TAGs, relevant national and international standards, and Relevant Good Practice (RGP) informed from existing practices adopted on nuclear licensed sites in Great Britain (Ref. 2, Ref. 6). The key SAPs and any relevant TAGs, national and international standards and guidance are detailed within this section. RGP, where applicable, is cited within the body of the assessment.

2.4.1 Safety Assessment Principles

19. The SAPs constitute the regulatory principles against which ONR judge the adequacy of safety cases (Ref. 2). The key SAPs applied within my assessment were SAPs ECH.1, ECH.2, ECH.3, ECH.4, EKP.1, SC.4, SC.5 and FA.1. A full list of SAPs applicable to Chemistry is included within Annex 1 of this report.

2.4.2 Technical Assessment Guides

20. The following TAGs were used as part of this assessment (Ref. 6):

- NS-TAST-GD-088 Chemistry of Operating Civil Nuclear Reactors
- NS-TAST-GD-089 Chemistry Assessment
- 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-005 ONR Guidance on the Demonstration of ALARP
- NS-TAST-GD-006 Design Basis Analysis
- NS-TAST-GD-007 Severe Accident Analysis

2.4.3 National and International Standards and Guidance

21. The following standards and guidance were used as part of this assessment:

- International Atomic Energy Agency (IAEA) Specific Safety Guide SSG-13, Chemistry Programme for Water Cooled Nuclear Power Plants (Ref. 9)
- IAEA Specific Safety Guide SSG-28 Commissioning for Nuclear Power Plants (Ref. 10)
- IAEA Specific Safety Requirements SSR-2/1, Safety of Nuclear Power Plants: Design (Ref. 11)
- IAEA Specific Safety Requirements SSR-2/2, Safety of Nuclear Power Plants: Commissioning and Operation (Ref. 12)
- Western European Nuclear Regulators Association (WENRA), Reactor Safety Reference Levels for Existing Reactors (Ref. 13)

22. There are both IAEA standards and WENRA Reference Levels 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 has been

benchmarked against both IAEA and WENRA guidance at the time of publication (Ref. 2).

2.5 Use of Technical Support Contractors

23. 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.

24. Table 1 below sets out the areas in which I used TSCs to support my assessment. I required this support to provide additional capacity and access to independent advice and experience.

Table 1: Work Packages Undertaken by the TSC

Number	Description	Reference
1	An embedded TSC provided specialist experience and independent advice in the assessment of the influence of the operating chemistry on materials selection, relevant degradation threats and through-life degradation in UK HPR1000. The TSC also supported the assessment of commissioning chemistry.	ONR 632
2	Independent review of selected submissions on the systems associated with chemistry control in the primary circuit of UK HPR1000.	ONR 634 Workpack 2
3	Independent review of the chemistry and radiochemistry sampling and monitoring systems in UK HPR1000.	ONR 634 Workpack 4
4	Independent review of the accident chemistry safety case for UK HPR1000.	ONR 634 Workpack 3

25. Whilst the TSCs undertook detailed technical reviews, this was done under my direction and close supervision. The regulatory judgment on the adequacy, or otherwise, of the generic UK HPR1000 safety case in this report has been made exclusively by ONR.

2.6 Integration with Other Assessment Topics

26. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot be carried out in isolation as there are often issues that span multiple disciplines. I have therefore worked closely with a number of other ONR inspectors and the Environment Agency to inform my assessment. The key interactions were:

- Assessment of the fuel design was led by Fuel and Core with Chemistry input on the effects of operating chemistry on cladding corrosion and fuel deposits.
- The Fault Studies and Severe Accidents assessments were led by their respective disciplines, with Chemistry input in areas where chemistry effects are important in determining the consequences or effectiveness of mitigation measures.

- Structural Integrity took the lead in assessing the case for the integrity of metallic components and structures. Chemistry covered the effects of the operating chemistry on susceptibility to material degradation mechanisms, including where integrity may not be threatened but corrosion is still important.
- The radioactive waste assessment was led by the Nuclear Liabilities discipline, with input from Chemistry on the assessment of chemistry control in the processing of liquid and gaseous wastes.
- Chemistry took the lead regarding normal operation source term assessment, leading a team of inspectors from other disciplines (Environment Agency, Radiological Protection and Nuclear Liabilities Regulation) to assess the adequacy of the normal operation source term.
- Chemistry led the assessment of the source term used in accident scenarios, alongside inspectors from Fault Studies and Severe Accident Analysis.

3 REQUESTING PARTY'S SAFETY CASE

3.1 Introduction to the Generic UK HPR1000 Design

27. The generic UK HPR1000 design is described in detail in the PCSR (Ref. 14). It is a three-loop PWR designed by CGN using the Chinese Hualong technology. The generic UK HPR1000 design has evolved from reactors which have been constructed and operated in China since the late 1980s, including the M310 design used at Daya Bay and Ling'ao (Units 1 and 2), the CPR1000, the CPR1000$^+$ and the more recent ACPR1000. The first two units of CGN's HPR1000, Fangchenggang Nuclear Power Plant Units 3 and 4, are under construction in China and Unit 3 (FCG3) is the reference plant for the generic UK HPR1000 design. The design is claimed to have a lifetime of at least 60 years and has a nominal electric output of 1,180 MW.

28. The reactor core contains zirconium clad uranium dioxide (UO_2) fuel assemblies and reactivity is controlled by a combination of control rods, soluble boron (in the form of boric acid (H_3BO_3)) in the coolant and burnable poisons within the fuel. The core is contained within a steel Reactor Pressure Vessel (RPV) which is connected to the key primary circuit components, including the Reactor Coolant Pumps (RCP), Steam Generators (SGs), Pressuriser (PZR) and associated piping, in the three-loop configuration. The design also includes a number of auxiliary systems that allow normal operation of the plant, as well as active and passive safety systems to provide protection in the case of faults, all contained within a number of dedicated buildings.

29. The Reactor Building (BRX) houses the reactor and primary circuit and is based on a double-walled containment with a large free volume. Three separate safeguard buildings surround the BRX and house key safety systems and the main control room. The Fuel Building (BFX) is also adjacent to the reactor and contains the fuel handling and short term storage facilities. Finally, the Nuclear Auxiliary Building (BNX) contains a number of systems that support operation of the reactor. In combination with the emergency diesel generator, personnel access and equipment access buildings, these constitute the Nuclear Island (NI) for the generic UK HPR1000 design.

3.2 The Generic UK HPR1000 Safety Case

30. In this section I provide an overview of the chemistry aspects of the generic UK HPR1000 safety case as provided by the RP during GDA. Details of the technical content of the documentation and my assessment of its adequacy are reported in the subsequent sections of my report.

3.2.1 Safety Case Structure

31. The generic safety case for UK HPR1000 follows a defined claims, arguments and evidence structure. The high-level safety claims are set out in 'PCSR Chapter 1 Introduction' and supporting 'chapter level' claims and arguments specific to chemistry are set out in 'PCSR Chapter 21 Reactor Chemistry' (Ref. 15, Ref. 5). The main chemistry claim is:

"Claim 3.3.10: The chemistry aspects of the plant design have been developed to reduce the nuclear safety risk ALARP."

32. A series of sub-claims also sit below the main chemistry claim.

33. The RP presents evidence in support of the chemistry claims, arguments and sub-arguments in a series of reports that support PCSR Chapter 21, as described below in sub-section 3.2.2. I make reference to the relevant claims and arguments in my assessment in Section 4 of this report and form a judgement on their adequacy and substantiation.

3.2.2 Safety Case Documentation

34. A description of the RP's chosen documentation hierarchy for the UK HPR1000 chemistry safety case is provided in 'Production Strategy for Reactor Chemistry' and is summarised in Figure 1 (Ref. 16). The overall documentation structure is divided into four tiers which are, in summary:

- Tier 1 comprises chapter 21 of the PCSR, which the RP considers to be the top level overview document for the chemistry safety case (Ref. 5). The PCSR chapter presents and develops the chemistry-related claims and arguments and is supported by detailed documentation presenting the underlying evidence.
- Tier 2 documentation is used to support and substantiate the claims in the PCSR chapter and provide the link between it and further detailed evidence presented in Tier 3 and Tier 4 documentation. The documents include the 'ALARP Demonstration Report of PCSR Chapter 21', an extensive series of topic reports covering specific areas such as commissioning chemistry and impurity control, accident chemistry submissions, System Design Manuals (SDMs) and other supporting reports (Ref. 17).
- Tier 3 submissions include detailed evaluation and analysis documents which aim to provide further support and substantiation to the chemistry claims and arguments set out in the PCSR and Tier 2 documentation. Tier 3 submissions include a series of component materials selection and ageing and degradation reports, Design Substantiation Reports (DSRs) providing the justification for the capability and availability of systems involved in chemistry control, and a series of documents containing the detailed accident chemistry analysis.
- Tier 4 submissions include the 'Production Strategy for Reactor Chemistry' and RQ responses (Ref. 16).

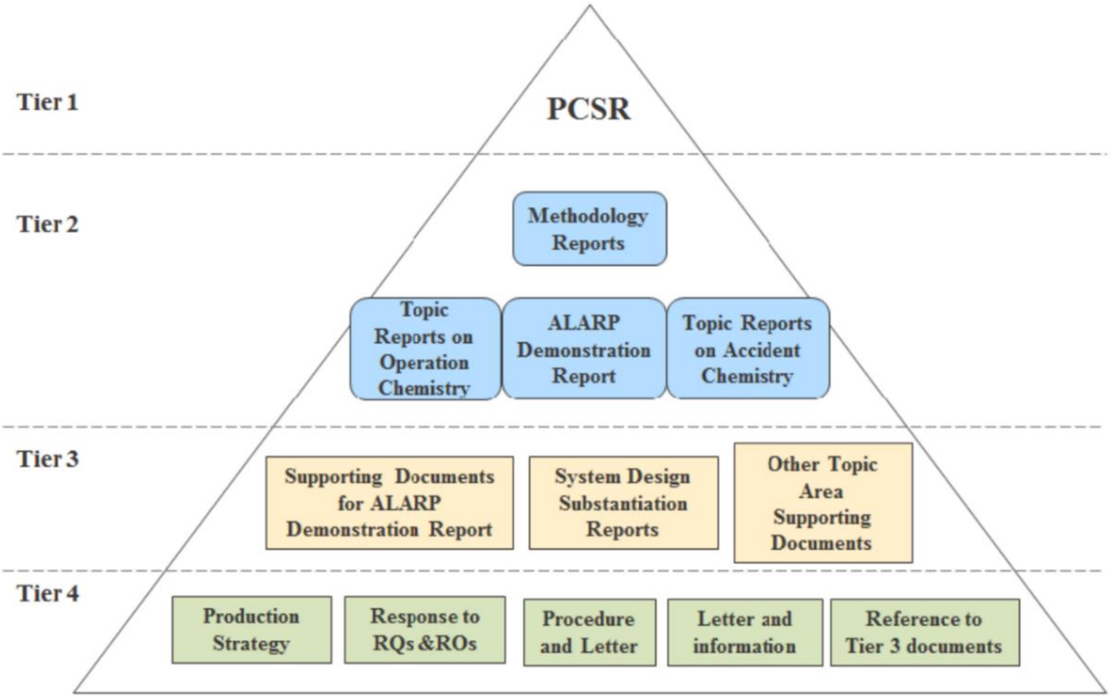

Figure 1: Documentation hierarchy of the UK HPR1000 chemistry safety case **(Ref. 16)**

35. The documentation forming the chemistry safety case is too extensive to list in full in this report, however it is set out in the 'Master Document Submission List' (MDSL) and key submissions are referenced and described in my assessment in Section 4 (Ref. 18).

3.2.2.1 Pre-Construction Safety Report (PCSR)

36. The PCSR is a top level document in the UK HPR1000 generic safety case which, together with the Pre-Construction Environmental Report (PCER) and the Generic Security Report, aims to demonstrate that the design meets UK safety, security and environmental requirements and that relevant risks are reduced SFAIRP. The PCSR consists of 33 chapters covering a general description of the plant, generic site characteristics and design principles, and chapters dedicated to key systems and technical topics.

37. Chapter 21 of the PCSR covers the chemistry aspects of the generic safety case (Ref. 5). The chapter describes the RP's chosen chemistry regime and presents the justification for why it considers that the selected chemistry is optimised and aligned with RGP. It considers the chemistry-related systems and seeks to substantiate their ability to maintain chemistry and radiochemistry parameters within the defined limits and conditions necessary in the interests of safety. The chapter is divided into the following topic areas:

- Primary water chemistry and associated systems
- Secondary water chemistry and associated systems
- Auxiliary water chemistry and associated systems
- Accident chemistry and associated systems
- Sampling and Monitoring

38. The chapter also draws a conclusion that the chemistry aspects of the generic plant design have been developed to reduce nuclear safety risks SFAIRP. A supporting report 'ALARP Demonstration Report of PCSR Chapter 21' presents further evidence in support of this claim (Ref. 17).

39. Several other chapters of the PCSR are of relevance to the scope of my assessment, including:

- Chapter 6 Reactor Coolant System (Ref. 19)
- Chapter 7 Safety Systems (Ref. 20)
- Chapter 10 Auxiliary Systems (Ref. 21)
- Chapter 11 Steam and Power Conversion System (Ref. 22)
- Chapter 13 Design Extension Conditions and Severe Accident Analysis (Ref. 23)
- Chapter 17 Structural Integrity (Ref. 24)
- Chapter 22 Radiological Protection (Ref. 25)
- Chapter 23 Radioactive Waste Management (Ref. 26)
- Chapter 28 Fuel Route and Storage (Ref. 27)

3.2.2.2 Topic Reports and Other Supporting Submissions

40. As noted in sub-section 3.2.2, Tier 2 documentation is used to support and substantiate the claims in PCSR Chapter 21 and provide the link between it and the further detailed evidence and analysis presented elsewhere in the generic chemistry safety case. Key Tier 2 submissions for the purposes of my assessment are the numerous topic reports on operating chemistry and accident chemistry. Of these, 'Topic Report on Power Operation Chemistry' and 'Topic Report on Start-up and Shutdown Chemistry' are used to summarise, at a high level, the information on chemistry during power operations and during start-up and shutdown that is contained in other more detailed Tier 2 submissions (Ref. 28, Ref. 29). The topic reports generally present the objectives and justification for chemistry controls in a particular area of the plant, or for a particular parameter, and consider the substantiation of the systems which control the chemistry, in conjunction with the relevant DSR.

41. Other important Tier 2 documents include the ‘Generic Water Chemistry Specification (LCO)’ and ‘Radiochemistry Parameters Value’ which summarise the chemistry and radiochemistry limits and conditions necessary in the interests of safety proposed for UK HPR1000 (Ref. 30, Ref. 31). The RP’s approach is to specify for each parameter a limit and an ‘operating window’ (expected range) for normal operation, in addition to classifying each parameter as either a ‘control’ or ‘diagnostic’ parameter. The RP defines control parameters as those parameters which play an important role in material integrity or fuel integrity and require strict control. It defines diagnostic parameters as those which assist chemistry staff in interpreting chemistry variations (Ref. 5).

3.2.2.3 Responses to Regulatory Queries and Regulatory Observations

42. As a result of my assessment of the generic safety case for UK HPR1000, I raised five ROs and 130 RQs (Ref. 4, Ref. 3). In ‘Production Strategy for Reactor Chemistry’ the RP states that the responses to the RQs and ROs were largely incorporated into the Tier 2 or Tier 3 documentation of the chemistry safety case (Ref. 16). I have considered the responses to ROs and RQs as part of my assessment and, as described in sub-section 4.8 of this report, I have considered the extent to which adequate consolidation of such information into the chemistry safety case has been achieved.

4 ONR ASSESSMENT

4.1 Structure of Assessment Undertaken

43. The following sub-sections describe the assessment undertaken for each of the main technical aspects identified in my Step 4 Assessment Plan and in Section 2 of this report (Ref. 7). I structured my assessment by system, focussing on chemistry control in the primary circuit, secondary circuit and key auxiliary systems including the SFP and the radioactive waste processing systems. Additionally, I structured my assessment of accident chemistry by fault category. Sections on my assessment of the demonstration that risks are reduced SFAIRP, and on the adequacy of the consolidated chemistry safety case, are also included.

44. Each section follows a similar structure:

- Firstly, a summary of the assessment is provided, outlining the relevant portions of the RP’s case being considered and the work undertaken to assess it. Some technical aspects may be divided into sub-topics, so the section may contain several sub-sections.
- Secondly, a summary of any key strengths and/or positive aspects of the safety case that are of relevance to the technical aspect is set out.
- Thirdly, as appropriate, a record of how any matters which have not been resolved on GDA timescales have been taken forward as Assessment Findings.
- Finally, the main conclusions of the assessment of the technical aspect are provided.

4.2 Primary Circuit

4.2.1 Overview

45. The Reactor Coolant System (RCP [RCS]) is the main system in the primary circuit. The RCP [RCS] consists of the RPV, SG, PZR, Reactor Coolant Pump (RCP) and Main Coolant Line (MCL). There are three loops of the RCP [RCS] linked to the RPV. The PZR is connected to the hot leg of the third loop through the surge line. Each loop consists of one SG, one RCP, and reactor coolant pipes (Ref. 5, Ref. 19). A simplified diagram of this arrangement is shown below in Figure 2:

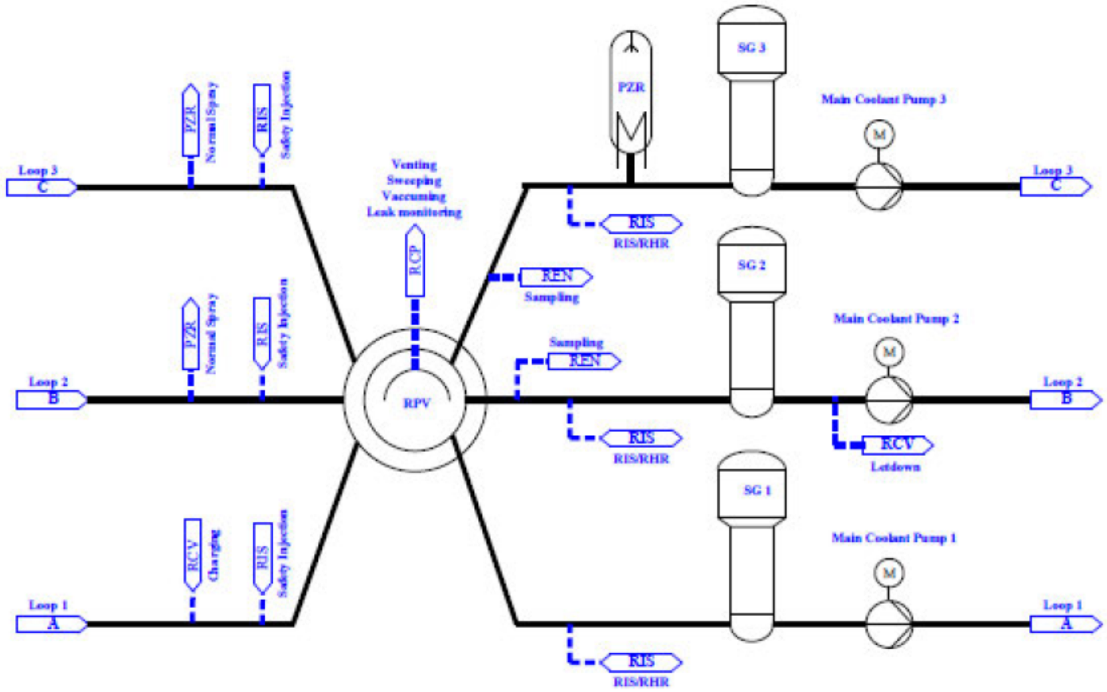

Figure 2: Simplified diagram of the Reactor Coolant System Main Loops

46. The primary circuit chemistry as specified by the RP is as follows:

- A target $pH_{300\,°C}$ of 7.2 as soon as possible following start-up. This is achieved with a lithium hydroxide (LiOH) concentration of up to 3.5 mg kg^{-1} at the Beginning of Cycle (BOC) and the use of Enriched Boric Acid (EBA). Where pH is referred to in this sub-section, it should be assumed to refer to $pH_{300\,°C}$, unless otherwise stated. This is assessed in sub-section 4.2.2.
- A target operating window for hydrogen of between [redacted] and 50 $cm^3\,kg^{-1}$. Throughout the remainder of this report, the units of hydrogen refer to cm^3 of hydrogen gas per kg of coolant at standard temperature and pressure conditions of 0 °C and 100 kPa. This aspect is assessed in sub-section 4.2.7.
- Impurities, such as chloride, sulphate and fluoride are minimised to a target of less than 150 µg kg^{-1} (Ref. 32). This aspect is assessed in sub-section 4.2.2.
- Zinc, in the form of zinc acetate ($Zn(CH_3CO_2)_2$) is added to the circuit, with a target operating concentration of between [redacted] and [redacted] µg kg^{-1}. This aspect is assessed in sub-section 4.2.6.

47. Corrosion processes in a PWR can be minimised by the design of the reactor, appropriate materials choices and selection of the surface finish which is applied to the surfaces of materials that are in contact with the coolant. These aspects have been developed over many years of plant operation and can lead to significant reductions in corrosion compared to older designs. Some corrosion is however inevitable, and the production of activated Corrosion Products (CPs) is the source for much of the radioactivity that occurs outside of the core during shutdowns. In order to minimise radiation exposure to operators, this should be minimised SFAIRP and understood in the safety case.

48. In the primary circuit, the two components which have the greatest surface area are the SGs and the fuel cladding. However, improvements in the materials used in the SGs and the cladding mean that they contribute increasingly smaller amounts of CPs compared to other parts of the primary circuit; this is discussed further in sub-section 4.2.3.1.

49. As well as corrosion of the primary circuit materials, corrosion of the materials of the systems that are attached to the primary circuit is also an important aspect of overall corrosion minimisation. Such systems include the Chemical Volume and Control System (RCV [CVCS]), the Reactor Boron and Water Makeup System (REA [RBWMS]), and the Coolant Storage and Treatment System (TEP [CSTS]).

50. In the UK HPR1000, a number of other systems support the RCP [RCS] in terms of chemical control, sampling and safety functions. The main systems are as follows:

- RCV [CVCS]
- REA [RBWMS]
- TEP [CSTS]
- Safety Injection System / Residual Heat Removal System (RIS [SIS] / RHRS)
- Emergency Boration System (RBS [EBS])
- Nuclear Sampling System (REN [NSS])

51. A diagram that shows how these systems interface is shown in Figure 3.

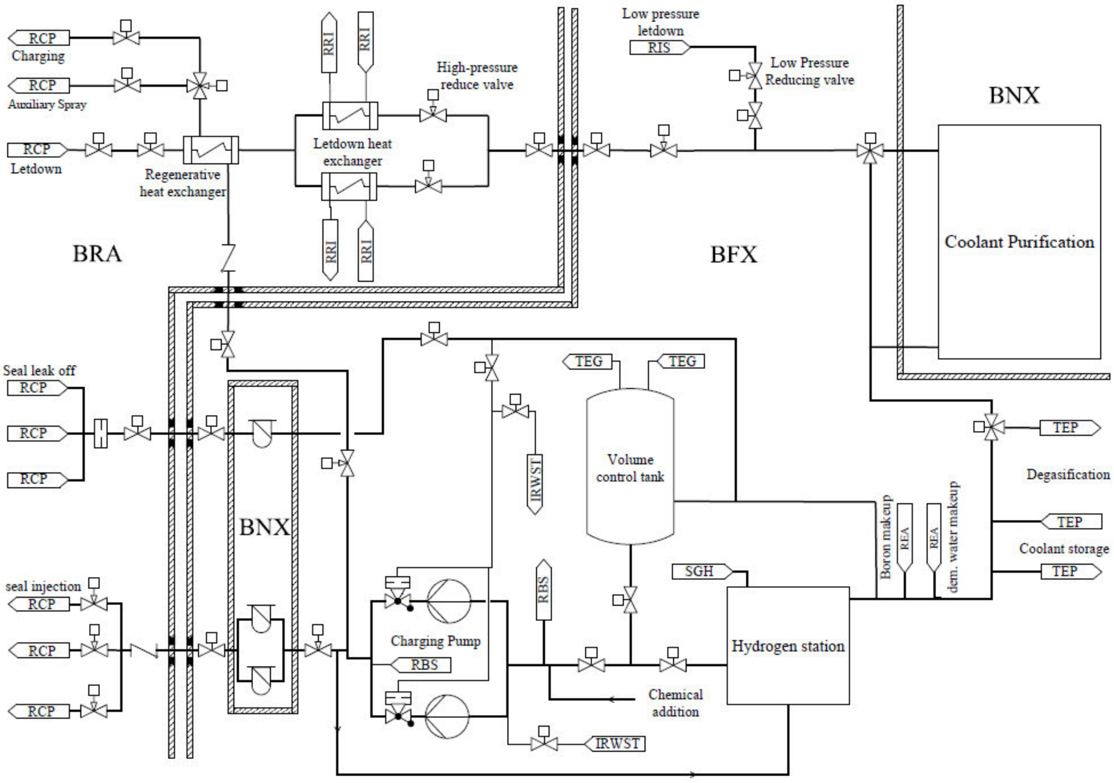

Figure 3: RCV [CVCS] Flow Diagram **(Ref. 33)**

52. The main systems that control the coolant chemistry are the RCV [CVCS], which includes the RBWMS, and the TEP [CSTS], and is assessed in sub-section 4.2.2. The RCV [CVCS] is supported by the REN [NSS] (described in sub-section 4.2.8) and the Component Cooling Water System (RRI [CCWS]) (described in sub-section 4.4.2).

53. Several safety systems form part of the primary circuit, and those of relevance to this part of the assessment are (Ref. 14):

- The RIS [SIS], which provides borated water injection into the RCP [RCS] to compensate for water inventory loss under certain Design Basis Conditions (DBCs). In addition, the RIS [SIS] can operate in Residual Heat Removal (RHR) mode, performing residual heat removal functions in the long-term after an accident.

- The RBS [EBS], which under certain DBCs injects highly borated water into the RCP [RCS] via the RIS [SIS] cold leg injection line to control the reactivity of the reactor during the transition from the controlled state to the safe state.

54. For the assessment of the primary circuit chemistry, the approach has been to sample several interrelated topics, including control of coolant chemistry, minimisation of corrosion and minimisation of radioactivity. The areas sampled were as follows:

- Chemical control
- Material, chemistry and radioactivity
- Material integrity and corrosion control
- Fuel integrity and fuel deposit formation
- Zinc addition
- Hydrogen dosing
- Control of safety system chemistry
- Sampling
- Overall approach to primary circuit chemistry
- Start-up and shutdown chemistry
- Commissioning and Hot Functional Testing (HFT)

55. My assessment of these topics is described in the following sections of this assessment report.

4.2.2 Chemical Control

56. This section of the report assesses the capability of the chemical control systems that support the plant, and their ability to provide the stated performance and reliability requirements. In the UK HPR1000, the main system that is used to control the chemistry of the primary circuit is the RCV [CVCS], and this section concerns the RCV [CVCS] during normal at-power operations. The role the RCV [CVCS] plays in start-up and shutdown chemistry control is assessed in sub-section 4.2.10. The RCV [CVCS] is supported by two other systems, which are closely involved in the control of chemistry in the primary circuit:

- The REA [RBWMS], which is responsible for the preparation and distribution of new boric acid solutions for all boron-containing systems.
- The TEP [CSTS], which is responsible for recycling enriched boric acid solutions via distillation, removal of gases from the primary circuit, and the removal of CPs from the primary circuit by the purification unit.

57. The RCV [CVCS] is designed to provide the following chemistry control functions (Ref. 33):

- Removal of CPs, fission products and some activation products from the reactor coolant.
- Removal of excess lithium ions from the reactor coolant and the injection of lithium hydroxide into the primary circuit to control the primary pH.
- Injection of hydrazine (N_2H_4) into the primary circuit for deoxygenation during plant start-up operation.
- Injection of hydrazine into the primary circuit for deoxygenation of the RIS [SIS] RHRS during plant shutdown operation.
- Addition of hydrogen into the reactor coolant to keep the dissolved hydrogen concentration at the required value.
- Injection of hydrogen peroxide (H_2O_2) for chemical dehydrogenation (for removing the dissolved hydrogen prior to oxygenation) and for the primary coolant oxygenation at temperatures lower than 80 °C to favour dissolution and purification of CPs.

- Injection of zinc (as zinc acetate) into the primary circuit for radiation field and stress corrosion cracking control.

58. As the RCV [CVCS], REA [RBWMS] and TEP [CSTS] are the main systems to control the primary circuit chemistry, a number of sub-claims are described which support the main chemistry claim “3.3.10 – the chemistry aspects of the plant design have been developed to reduce the nuclear safety risks ALARP”, as follows:

- SC21.1 – the primary chemistry and process are optimised in all operating modes to maintain the integrity of the safety barriers in the primary circuit.
- SC21.2 – the chemistry regime supports the control of reactivity in all operating modes.
- SC21.3 – radioactivity in the primary circuit is reduced SFAIRP to minimise worker and public dose.

59. Assessment of the chemical control systems in the UK HPR1000 began in Step 4 of GDA. I commissioned a TSC to support this important area during Step 4, and the TSC was tasked with assessing all aspects of the operation of the chemical control systems (Ref. 34). Although I did not sample these systems in detail during Step 3 of the GDA project, my assessment did identify that the generic safety case at that stage did not include an adequate description of the control of boron chemistry, which is a fundamental requirement of these systems. As such, I raised RO-UKHPR1000-0031, part of which required the RP to adequately justify the control of boron chemistry in the UK HPR1000 (Ref. 4). My TSC contributed to the assessment of the closure of RO-UKHPR1000-0031, which I will describe in further detail in sub-section 4.2.2.

60. At the beginning of Step 4 of GDA, I identified the following aspects for sampling:

- Adequate justification of system functional requirements.
- Chemical addition, with a particular focus on hydrogen addition.
- Control of boron chemistry (concentration and enrichment) and boron dilution prevention.

61. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment, and provide guidance on a wide range of expectations when assessing the safety case for operating civil nuclear plants (Ref. 6).

4.2.2.1 Chemical Volume and Control System

62. The purification unit of the RCV [CVCS] (Figure 4), which consists of resin demineralisers and filters, is used for continual purification during power operation to remove fission, corrosion and activation products in order to control the radioactivity level in the primary circuit. The purification system also removes, and is an aid in the management of, non-activated chemical impurities, such as sodium and chloride.

63. To achieve this function, the RCV [CVCS] receives primary coolant from the RCP [RCS], via the letdown line, where a regenerative heat exchanger and letdown heat exchanger reduce the temperature of the coolant. The pressure is reduced via the high pressure reducing valve before the cooled letdown is directed to the purification system of the RCV [CVCS]. The letdown is initially filtered through a replaceable filter cartridge to remove suspended solids before entering a mixed bed resin demineraliser, removing suspended solids and ionic species. Typically, only one of two available demineralisers are in operation at any one time, affording a level of redundancy in the system. The flow leaving the mixed bed demineraliser passes through a resin filter and

intermittently enters the cation bed where lithium removal is used for pH control. The RP clarified, in response to RQ-UKHPR1000-1002, that the non-lithiated cation bed can be brought online in the event of failed fuel for additional fission product removal capacity (Ref. 3).

64. The purified letdown is sent ([redacted]% of the flow) via the hydrogenation and chemical injection systems, before passage through the charging pumps and regenerative heat exchanger. The letdown flow is injected to the primary circuit by the charging pumps with the charging line and the seal injection line. A proportion of the letdown flow ([redacted]%) is sent to the Volume Control Tank (VCT) to ensure the boron concentration remains consistent with that of the RCP [RCS]. The purified letdown can also be routed through the TEP [CSTS] for degasifying and boron recovery purposes, which is described in more detail in sub-section 4.2.2.3. The RP states that the technology outlined for the UK HPR1000 has been applied in operating CGN plants. Based on this previous experience, it is reported that the chemistry and radiochemical parameters can be controlled effectively (Ref. 33).

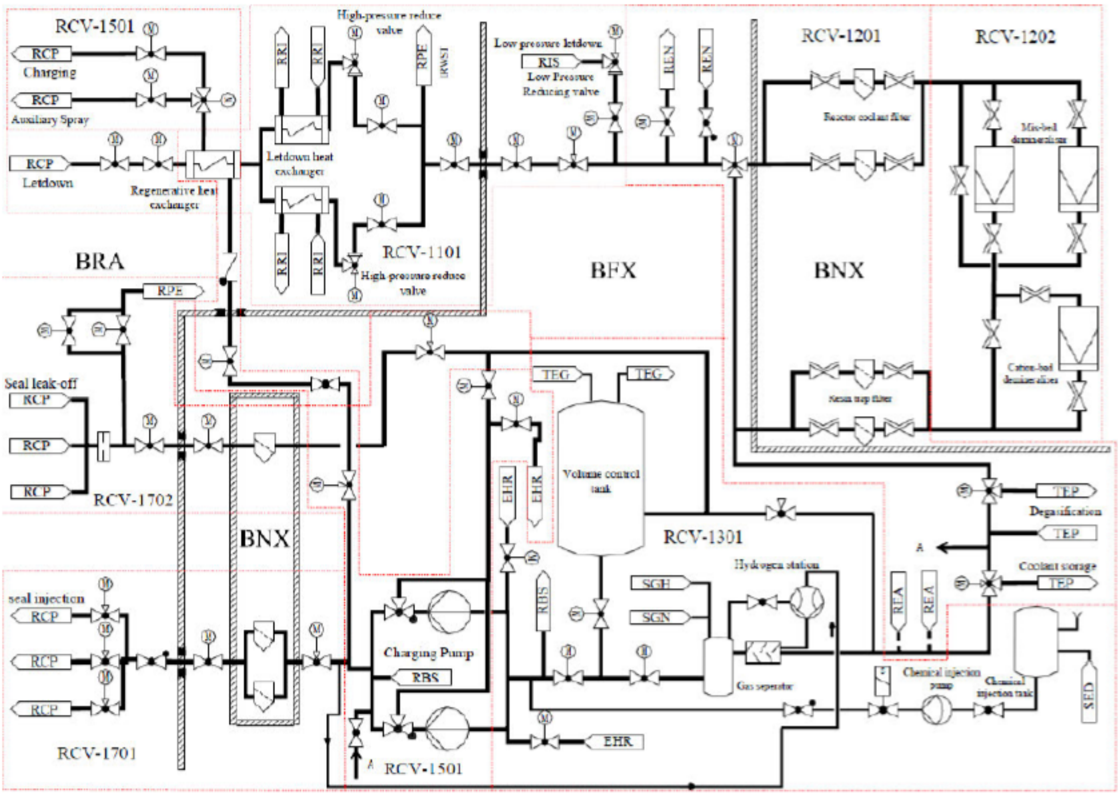

Figure 4: Simplified Diagram of the RCV [CVCS] **(Ref. 33)**

Ion Exchange

65. The RCV [CVCS] demineralisers play an important role in the control of primary circuit chemistry and represent a crucial element in the generic UK HPR1000 design to fulfil the outlined chemistry functions. One significant aspect of the demineraliser which needs to be considered is the system capacity; ensuring the generic design has taken account of the species present, the operational demands and the duration over which it must operate. The two mixed-bed demineralisers, saturated with lithium and borate ions, are primarily to remove ionic impurities from the primary coolant. The single cationic bed (hydrogen form) controls lithium concentrations within the primary coolant and also removes cationic impurities such as caesium resulting from fuel clad defects.

66. The RCV [CVCS] DSR, outlines that the resins have capacity for an 18-month cycle which was based on a series of reported theoretical calculations (Ref. 33). This approach was queried, and additional evidence sought to substantiate the values

reported. The RP reported that the quoted capacity is based on CPR1000 Operating Experience (OPEX) where the average resin replacement cycle is typically [redacted] years (mixed bed) and [redacted] years (cation bed). Additionally, there are two mixed bed demineralisers, one in operation and one in stand-by. While a level of redundancy for an 18-month cycle has been outlined for the UK HPR1000, with the ability to instate a second demineraliser should the need arise, no supporting operational data was provided by the RP in response to RQ-UKHPR1000-1292 (Ref. 3).

67. My expectations for the safety case during GDA are that where relevant, plant data from similar plants should be available to support the claims and arguments that the RP makes; during Step 4 of GDA, very little OPEX was included in the generic safety case. I therefore consider this to be an Assessment Finding.

AF-UKHPR1000-0011 – The licensee shall, as part of developing the operational chemistry safety case, ensure robust operational plant data is included to support safety case claims and arguments. A methodology should be implemented to identify, select and use relevant data to underpin the safety case. The data should be used to demonstrate, amongst other matters, that the following are appropriate and achievable:

- plant operability;
- control of chemistry parameters;
- representative sampling of relevant chemical species; and
- limits and conditions.

68. The use of OPEX in the safety case was raised as RO-UKHPR1000-0044 during Step 4 of GDA (Ref. 4), which identified gaps in the:

- approach to identifying, justifying and using OPEX;
- scope (depth and breadth) of the OPEX identified and selected; and
- what OPEX (including data) is presented in the generic safety case, and how.

69. Chemistry was one of the topics in which OPEX is important, and chemistry was considered as part of this cross-cutting RO. Several topic areas have raised concerns about the presentation of plant data within the safety case. The scope of RO-UKHPR1000-0044 was purposefully different to what the individual disciplines have assessed in terms of the adequacy of OPEX data with respect to its use in supporting topic specific claims and arguments. The topic of OPEX is a repeating one throughout this report, and I will discuss further examples in later sub-sections.

70. My TSC considers that for PWRs, a typical resin replacement frequency is likely to mean a change of resins every operating cycle and noted that the suggested resin replacement frequency of between [redacted] and [redacted] years may not represent a position in which risks are reduced SFAIRP in terms of source term, noting the accumulated activity that such a replacement frequency could result in its report (Ref. 34). Whilst it is important that the function of the RCV [CVCS] should be adequately justified in all aspects, the design of the RCV [CVCS] will be confirmed by the licensee, and an adequate justification of its functions should be conducted as part of normal business.

71. The RCV [CVCS] DSR, reports that if the cation bed should ‘fail’ (assumed to refer to exhausted resin), then the lithium concentration can be reduced through the TEP [CSTS] and REA [RBWMS] (Ref. 33). This, together with the intermittent operation of the cation bed, is used to support the lack of redundancy. This appears appropriate based on the information presented, however I consider that OPEX in the form of data from a similar plant is required to support the claims about the cation concentration and

whether the level of cation redundancy is appropriate. This is part of the Assessment Finding AF-UKHPR1000-0011.

72. A further important aspect of the operation of the demineraliser is in response to a failed fuel event. In response to such an event; two charging pumps are placed into service, doubling the purification rate. If one of the mixed-bed demineralisers becomes exhausted, then the second is placed into service. The purified letdown flow can be discharged to the degasification unit of the TEP [CSTS] for removing fission gases. Because the lithium-form cation resin has little affinity for the fission product of caesium, the cation bed demineraliser with a hydrogen-form cation resin may be put into operation for removing caesium when needed. This action will also remove lithium, so lithium hydroxide additions by the RCV [CVCS] chemical addition unit may be needed to compensate removed lithium.

73. My TSC considered that the inclusion of two mixed beds and the cation bed provides sufficient redundancy against the additional loading as a result of failed fuel (Ref. 34). The TEP [CSTS] mixed resin bed (H^+ form) will also provide a degree of redundancy and additional clean-up although this has not been reported against failed fuel. I am therefore content that the arrangements for the purification systems are reasonable.

74. The RP has designated the function of controlling the main primary chemical parameters (which includes aspects of the operation of the RCV [CVCS], REA [RBWMS] and TEP [CSTS]) a categorisation of FC3 (Ref. 35). Considering the claims made in the generic safety case and safety functions provided by the RCV [CVCS], from a chemistry perspective, I am satisfied FC3 is an appropriate categorisation.

Filtration

75. The primary coolant filters play an important role in the removal of suspended solids, which can contribute to deposition throughout the primary circuit and on fuel elements (as fuel deposits). They can also impact the operability/life span of the resin demineralisers. Therefore, it is important to ensure that these filters are sized accordingly. The RP has provided literature particle distribution values to justify the mesh sizing of [redacted] µm. However, my TSC concluded that insufficient evidence had been provided by the RP to describe how representative this mesh size is for the UK HPR1000 (Ref. 34). Should an incorrect filter size be used, a greater proportion of solids would deposit onto the ion exchange resins, which could affect their functionality and require more frequent changes. I consider this to be a minor shortfall.

76. Following a request made in RQ-UKHPR1000-1002 by my TSC for the RP to provide details about the monitoring requirements placed upon the filters to ensure their operability and function, the RP described the monitoring of the differential pressure drop across the filters, radioactivity monitoring and the availability of stand-by filters which are changeable during operations (Ref. 3). My TSC considered that this provided adequate confidence in the continued safe operability of the filters. The approach to changing these filters, and any impacts this has on worker exposure to radiation has been assessed by Radiological Protection colleagues (Ref. 36).

77. The RP provided information about the ion exchange resin filters, stating that they are comparable to the those featured in the CPR1000 design, where greater than 98% efficiency is achieved (Ref. 34). The RP confirmed that resin diameter and fines distribution for the HPR1000 is comparable to the CPR1000, which provides confidence in the operability and suitability of the resin filter. However, no OPEX was provided to support these claims. This is part of Assessment Finding AF-UKHPR1000-0011.

4.2.2.2 Reactor Boron and Water Makeup System

78. The REA [RBWMS] is responsible for the preparation and distribution of new boric acid solutions, providing a solution with the appropriate concentration and boron-10 (^{10}B) abundance for all boron-containing systems, such as the Fuel Pool Cooling and Treatment System (PTR [FPCTS]), the RBS [EBS], and the RIS [SIS]. A description of its operation is provided in PCSR Chapter 21 (Ref. 5).

79. The design and process applied for the UK HPR1000 is reported to be the same as the reference plant, FCG3. Compared to the CPR1000, the management of boron for the UK HPR1000 is the same, except for the selection of EBA (assessed in sub-section 4.2.2.4), and the inclusion of an RBS [EBS], assessed in sub-section 4.5.2.1). The management of ^{10}B enrichment and the RBS [EBS] in the UK HPR1000 is described by the RP to be the same as the RP's operating plants.

80. Control of the boron chemistry is an important function of the REA [RBWMS]. Following clarification in response to RQ-UKHPR1000-1124, the RP reported that the flows are connected to an alarm and automatic interrupt, should the flow deviate from the prescribed value. Following distribution, the boric acid concentration and ^{10}B abundance are measured, which represents an alternative means in ensuring correct make-up has been conducted.

81. The RCV [CVCS] injects boric acid provided by the REA [RBWMS] into the primary circuit through the charging lines and discharges the excess reactor coolant to the TEP [CSTS] through the letdown lines. The RCV [CVCS] also undertakes part of the protection against accidental dilution, and isolates the VCT and the hydrogenation station from the suction of the charging pumps when the condition of incorrect dilution occurs.

82. The Boric Acid Storage Tank (BAST) is used to store the 4 wt.% boric acid to meet the demands for slow reactivity control and volume control during unit normal operation. Two BASTs are provided. One BAST is online to the TEP [CSTS] for accepting the recycled concentrated boric acid (4 wt.%), the other online to the RCV [CVCS] for normal make-up. The RP claims that the BASTs are sufficient to bring the core to a subcritical state in cold shutdown from power operation following refuelling shutdown, and a total of [redacted] m^3 boric acid (^{10}B enrichment at 35 at%) is required during unit normal operation; each BAST holds 70m^3. The RP clarified in response to RQ-UKHPR1000-1124 that [redacted] m^3 boric acid is only required following start-up following cold shutdown, (after a refuelling shutdown) (Ref. 3). I am content with the rationale provided, and considered the boron concentration monitoring on BAST feeds from the TEP [CSTS] to be adequate (Ref. 34). I am therefore content with the arrangements for the BASTs.

83. To ensure continued operability of the REA [RBWMS], the pump actuators and valves of the boric acid storage and injection subsystem, together with the demineralised water injection subsystem, are equipped with an emergency power supply, ensuring the REA [RBWMS] can also be put into operation in case of Loss Of Off-site Power (LOOP) supplies.

84. Boric acid crystallisation poses a risk to the functionality and operability of the REA [RBWMS], either through a reduction in concentration or as a result of blockages/pump malfunction. The intended use of EBA is reported to reduce this risk; however crystallisation may still occur under certain circumstances, notably at low temperatures. To mitigate this risk, it is common practice for solutions of boric acid to be heated to prevent crystallisation. I raised a query within RQ-UKHPR1000-1124 to clarify the mitigations the RP had employed to prevent crystallisation (Ref. 3). The RP stated that the pipes and buildings are maintained at or above 20 °C (an alarm signal is set at 20 °C), through the use of two convectors, each on two trains with emergency diesel

power if required. The safety case states that the concentration of boron (as boric acid) stored in the BAST is 7000 – 7700 mg kg^{-1} and the crystallization temperature at this concentration is about 17 °C (Ref. 37). Further clarification was provided by the RP in the response to RQ-UKHPR1000-1414, which stated that firstly the boration safety function is provided by the RBS [EBS] rather than the RBWMS. Secondly, the room in which the BASTs are located is maintained above 26 °C by convector heaters, with alarms at 26 and 23 °C. Should the heating fail, the boric acid injection pump can be used to heat the contents of the tank. And finally, OPEX from similar plants had not identified any incidents where boric acid had crystallised. I considered that the RP's responses provide sufficient confidence that the measures applied will prevent crystallisation (Ref. 3, Ref. 34).

85. The integrity of boron-containing systems is important not only in maintaining functionality but also to mitigate the risk of boric acid corrosion. Instances have been reported where small leaks of boron-containing solution have led to significant corrosion issues, notably the reactor pressure vessel head degradation incident at Davis-Besse nuclear power station in the US. In RQ-UKHPR1000-1124, I asked the RP to clarify the mitigations in place for the UK HPR1000 to prevent such an incident (Ref. 3). In response, the RP highlighted that the choice of corrosion resistant materials in the REA [RBWMS] (Stainless Steel and Alloy 690), and periodic In-Service Inspection (ISI), together with the leak monitoring system, were provided as mitigations. I expect leak management processes to be developed by the licensee, and as these systems operate at relatively low temperatures, I am content that the justification provided by the RP is an adequate position to reach at the end of GDA. The issue of boric acid corrosion also applies to other systems that handle boric acid, such as the RCV [CVCS] and the TEP [CSTS]. As these systems use the same materials as the REA [RBWMS], I am also content that these designs adequately mitigate the risk of boric acid corrosion.

4.2.2.3 Coolant Storage and Treatment System

86. The TEP [CSTS], consists of a purification unit, evaporator and a condensate degasification unit which are used to meet the chemistry and radiochemistry control requirements. The chemistry control requirements for the TEP [CSTS] during normal operation (including power operation, start-up and shutdown operation) are to ensure that:

- The boron content of the concentrated boric acid produced by the evaporator unit is controlled between 7000-7700 mg kg^{-1}.
- The oxygen content of the treated demineralised water is no more than 0.05 mg kg^{-1}.
- The coolant degasification unit can be used during shutdown to remove fission gases (notably noble gases) dissolved in the primary coolant.
- The unit removes solid and ionized impurities, so as to reduce the radioactivity of the reactor coolant to be treated.

87. The purification system of the TEP [CSTS] receives reactor coolant from the RCV [CVCS] and Nuclear Island Vent and Drain System (RPE [VDS]). Purification is conducted by use of a mixed bed demineraliser which is reported by the RP to be the same ion exchange technology applied at Daya Bay power plant and that the technology is mature and reliable.

88. The resin bed receives coolant which has already seen treatment by the RCV [CVCS] purification unit so impurity levels should be minimal. The TEP [CSTS] mixed bed resin will provide an additional measure to ensure impurities and radiochemical species are further reduced, preventing redistribution in the boric acid concentrate. The presence of a resin filter should prevent the transport of resin fines throughout the system. My TSC considered that the sizing of the resin bed appears appropriate against the range

of species reported (Ref. 33, Ref. 34). However, the calculations do not consider the lithium loading to the cation bed. Following a request for clarification made in RQ-UKHPR1000-1511, the RP confirmed that the flow to the bed is not always routed via the RCV [CVCS] cation bed (Ref. 3). Based on this, it would appear that the sizing calculation has not considered a potentially significant lithium contribution which may result in a lack of capacity and impurity/lithium carryover with the boric acid feed, ultimately contaminating the BAST. Additionally, no detail of the resin replacement strategy has been provided, which may strengthen the case. I consider this to be a minor shortfall because it can be dealt with by more frequent resin changes.

89. Following treatment by the resin demineralisers, the TEP [CSTS] utilises an evaporation process to prepare demineralised water and a 7000-7700 mg kg^{-1} boric acid solution. It is reported by the RP that this thermal separation technology is reliable and used widely. While no CGN experience has been provided for the use of this technology, the RP reports it to be used at EPR™ nuclear power plants (Ref. 33). My TSC considered that the relatively simplistic nature of the technology together with the application elsewhere provides a reasonable assurance that this aspect of the TEP [CSTS] will be able to provide the required function, and I am therefore content that the arrangements are adequate for GDA (Ref. 34).

90. The TEP [CSTS] Design Justification Report (Ref. 33), reports that there are two degasification units in the TEP [CSTS]; a coolant degasification unit and distillate degasification unit. The technology employed is the application of a vacuum at 50 °C whilst passing through a column, where the gases are extracted through the vacuum pump, and the RP states that this technology is employed at EPR™ nuclear power plants (Ref. 33). Consideration has been given to the requirements placed on the system, namely the removal of gaseous fission products and dissolved oxygen. However, the RP did not provide operating experience to support the use of this technology which would provide further support to the intended design. This is part of Assessment Finding AF-UKHPR1000-0011.

91. The six coolant storage tanks are reported to be constantly swept with nitrogen and maintained at a reduced pressure. This provides assurance that the accumulation of hydrogen and fission products is minimised, with the reduced pressure and gas sweeping preventing hydrogen from escaping from the tanks.

4.2.2.4 Boron Control

92. The control of boron chemistry, both its concentration and its enrichment, is a key requirement, and should be an important feature of the safety case. Boron, in the form of boric acid, is dissolved in the primary coolant of PWRs to control the core power level during normal operation, and by acting as a backup to the control rods when the reactor is shutdown. Stocks of coolant containing boric acid are required to be stored for use during emergencies, at shutdown and for use in the SFP. The precise enrichment of the boron required, and its concentration at any one time, are determined by nuclear physics. The operators of any PWR must have rigorous control over the boron chemistry, as it is often one of the two controls employed to prevent unintentional nuclear reaction. During Step 3 of GDA, my initial assessment in this area identified a lack of information about how boron chemistry was controlled, and there was very little description of how the interaction of the different items of plant that are involved with the control of boron chemistry occurs. I also identified that there was a lack of an adequate description of boron dilution faults. During Step 3 of GDA, it was also apparent that although the RP intended to use EBA in the design, there was no consideration or justification of this in the safety case, and I therefore raised RO-UKHPR1000-0031 (Ref. 4).

Description of the Boron Cycle and System Overview

93. The first of two actions of RO-UKHPR1000-0031 (the second is related to boron dilution faults, assessed later in this sub-section) required the RP to demonstrate that the boron chemistry can be adequately controlled during normal operations. In response to this, the RP provided two documents. The first justified the use and control of EBA, which I have assessed earlier in this sub-section. The second document, ‘Boron Management and Risk Evaluation during Normal Operation’, provided a description of the different phases of the life cycle of boron in the plant, namely: boric acid preparation and distribution, boric acid storage, boron adjustment, boron transfer, boron recovery for reuse and boron discharge (Ref. 37).

94. The Boron Management and Risk Evaluation report describes how during normal operation, the boron concentration in the primary circuit is gradually diluted by the injection of demineralised water into the RCP [RCS] (Ref. 37). This is to maintain reactivity, due to fuel burnup during power operation. This process is performed via the REA [RBWMS] via the RCV [CVCS].

95. Following a description of the systems involved in the control of the boron chemistry, the Boron Management report goes on to detail the risks that the RP has identified in the operation of the relevant systems, and the countermeasures that have been adopted to mitigate those risks (Ref. 37). The ability of the plant to deliver the required functions and chemistry have been assessed by my TSC, and are described earlier in this section. This review considered how the concentration and enrichment are controlled throughout the cycle. However, in terms of meeting the requirements of RO-UKHPR1000-0031 to provide a description of the control of the boron chemistry in all operating modes, I judged that the RP provided sufficient information.

Use of Enriched Boric Acid (EBA)

96. The reference design, and reference plant, of UK HPR1000 uses EBA (enriched in ^{10}B) rather than natural boric acid. Historically, many PWRs used natural boric acid, but more recent plants have seen increased usage of EBA. The use of EBA has benefits for aspects of reactor operation (such as supporting an optimum pH in the primary circuit to minimise corrosion). However, the safety case that was presented in Step 3 of GDA did not provide sufficient justification of the use of EBA, nor how the use of EBA would be controlled. I therefore raised RO-UKHPR1000-0031 which, amongst other actions, required the RP to justify the use of EBA and the level of enrichment (Ref. 4).

97. The RP provided a justification of the use of EBA, and a target enrichment level to be used in the primary circuit of the UK HPR1000 (Ref. 38). As referred to previously, the use of EBA allows a lower concentration of boric acid to be used than would be the case for natural boric acid. This results in the need for lower concentrations of lithium hydroxide to maintain the target pH of 7.2 that the RP has chosen (see sub-section 4.2.3). The RP initially claimed that a lower lithium concentration would result in lower tritium (^{3}H) generation but did not provide any details. In response to RQ-UKHPR1000-1003, the RP stated that the average lithium concentration for an EBA system compared to a natural boric acid system would be 1.6 mg kg^{-1} compared to 2.6 mg kg^{-1} (Ref. 3). This would result in approximately 1.2 TBq yr^{-1} less ^{3}H. The use of EBA also means that the target pH can be reached more quickly at the BOC, than would otherwise be the case; high lithium concentrations are not permitted due to concerns over corrosion of the fuel cladding material. The scoring system devised by the RP to demonstrate why the choice of EBA reduces risks SFAIRP reflects these positive aspects, whilst also highlighting increased costs with the use of EBA (Ref. 38).

98. In order to recover the valuable EBA, the design of the UK HPR1000 includes the requirement to recycle the primary coolant. This introduces detriments in terms of the potential to accumulate impurities and radioactivity. To do this, discharged coolant will

undergo evaporation in the TEP [CSTS], where EBA will be recovered from the evaporator concentrate and returned to the primary circuit via the REA [RBWMS]. My assessment of the TEP [CSTS] can be found in sub-section 4.2.2.3.

99. In response to RQ-UKHPR1000-1003, in which I asked how these aspects were considered in the evaluation of the design, the RP confirmed that although the UK HPR1000 will use EBA, like the reference design but unlike the CGN fleet of reactors, the process systems in the design of the EBA plant are not significantly different to those that use natural boric acid (Ref. 3). The main difference is the concentration of boric acid, but this does not affect the flow route or the mechanism of boron recycling. Recycling of EBA may affect the production and transport of ^{3}H, and this is assessed in sub-section 4.2.3.3.

100. I also asked the RP in RQ-UKHPR1000-1003 about the detriments associated with any impurity and radioactivity accumulation, through the use of EBA (Ref. 3). The RP stated that the long-term recycling may lead to impurities and radioactivity accumulation in the BASTs and primary circuit. This factor is independent of the choice to use EBA, as the majority of plants using natural boric acid recycle the coolant to prevent discharge of boron from the plant (Ref. 37).The impurities and radioactivity in suspended solids or ionic species can be removed by filters and resin ion exchange in TEP [CSTS] and the RCV [CVCS] during normal operation. A more detailed review of the potential impact that the use of EBA may have on waste arisings was added to the Selection of Enriched Boric Acid for the UK HPR1000 (Ref. 38). This included consideration of the impact of EBA use on liquid and solid waste arisings, and generation of ^{3}H. This review concluded that as less boric acid is required to be used when using EBA compared to natural boric acid, overall total waste arisings are expected to be significantly less. Any arisings from coolant recycling are expected to be significantly less than reductions expected through the use of EBA (Ref. 38). I am content that the RP has considered these aspects adequately.

101. The RP also discussed the potential build-up of silica in the BASTs, as this species is particularly difficult to remove from the primary circuit, due to it being poorly removed by ion exchange resins and filters. There are a number of possible forms of silicates, but for the purposes of this report, they will be simplistically referred to as silica. The RP stated that the silica concentration will be monitored, and if the silica concentration exceeded the limits, then the concentrated boric acid in the BAST can be discharged via the RPE [VDS] at the bottom of the BAST, and it can be replaced by a freshly prepared batch of boric acid. The RP recognises that some discharge is inevitable, but I was reassured by their cognisance of the importance of maintaining high quality chemical feedstocks, to mitigate this as far as possible (Ref. 37).

102. The RP also describes the target enrichment level to be used in the UK HPR1000, which is based upon the core design. The range is selected to be between 35 and 39 at%, with a target of 37 at% (Ref. 38). This figure is similar to other plants that operate with EBA. I will assess the controls required to maintain this enrichment in the next sub-section of this report. Theoretically, a higher enrichment of boric acid would allow a higher operating pH in the primary circuit, which could have some benefits in terms of corrosion performance of the primary circuit materials. However, the RP argues that such an increase would affect fuel management and would represent a significant increase in costs, for a small effect in terms of corrosion. As the proposed operating pH and enrichment levels correspond with international good practice (see sub-section 4.2.3.3 on radioactivity), I am content that it is not appropriate for a higher enrichment of boric acid to be used. Any aspects of sampling and monitoring of the concentration of boron are contained within sub-section 4.2.8.

Control of Boron Enrichment

103. My assessment of boron dilution faults is set out later in this section, but when using EBA, the enrichment level of the boron must also be controlled to prevent reactivity control issues. The RP describes two mechanisms through which the EBA can become depleted in ^{10}B (Ref. 37):

- Loss of boron in the RCV [CVCS] and TEP [CSTS] demineralisers (through resin absorption).
- Consumption of ^{10}B through neutron absorption and subsequent decay to lithium-7 (^{7}Li) and an alpha particle (the ^{10}B (n, α) ^{7}Li reaction).

104. For losses via resin absorption, the RP states that once the anion resin in mixed bed demineralisers is put into operation for the first time or after the resin is replaced without being first saturated with BO_3^-, the reactor coolant would be diluted when flowing through the demineraliser. However, it should be noted that:

- Unlike the RCV [CVCS], the demineraliser in the TEP [CSTS] is arranged upstream of the evaporation unit, which does not affect the boron concentration at the outlet of the evaporation unit, so this part of boron loss will not affect the boron concentration in the RCP [RCS].
- The ion exchange media has no selectivity for different isotopes of boron, therefore the saturation does not affect the proportion of ^{10}B in the reactor coolant. However, inadvertent use of an incorrect resin (saturated with natural boron) could dilute the EBA.

105. The RP provided an estimation of the exchange capacity of the resins, in order to estimate the amount of boron that would be removed via this mechanism. To account for these losses, fresh boric acid is made up and added to the BASTs. I was content with this description.

106. The RP also provided an estimate of the depletion of ^{10}B during the fuel cycle from reaction with neutrons, which is the principle behind normal reactivity control. Starting with an enrichment of 37 at%, the RP estimates the abundance of ^{10}B in the primary circuit at the end of the cycle to be about 35.3 at%. During the cycle, with fuel burnup, the boron in the primary circuit is gradually recycled into the BAST. The RP estimates the abundance of ^{10}B in the BAST at the end of the cycle to be 36.5 at%. I note that both are within the target operating enrichment. This loss of ^{10}B is estimated to be equivalent to about 7 kg per cycle.

107. There are no safety consequences of the fall of enrichment during the cycle as long as the ^{10}B concentration is within limits. As fuel is burnt up, the requirement for the abundance of ^{10}B also falls, and the ^{10}B concentration can be maintained by reducing the dilution volume (therefore increasing the total boron concentration). This is a feature of all PWRs, and I am content with the explanation provided by the RP in this aspect of the resolution of RO-UKHPR1000-0031.

108. Control of boron enrichment is a fundamental aspect of the REA [RBWMS]; fresh boric acid solutions are made up in the system, for distribution to various other systems of the UK HPR1000, including the primary circuit and the In-containment Refuelling Water Storage Tank (IRWST). Highly enriched boric acid (up to 96 at%) can be added to the REA [RBWMS] should the enrichment level need to be increased. The RP describes how the boric acid requirements are calculated prior to make-up, and then before it is sent to another system the concentration and ^{10}B abundance are sampled and analysed to determine that the correct concentration and abundance have been achieved. Following a request for further information raised in RQ-UKHPR1000-1124, the RP confirmed that this process is manual, that a spreadsheet would be provided to operators to help with the process, and that operating rules and procedures would be

developed at the site specific stage (Ref. 3). I was content with this response, in terms of controlling enrichment during make-up, but it also indicated an approach that was largely manual rather than automatic. I therefore asked about the philosophy of the boron control processes, and how it had been decided to utilise manual rather than automatic processes.

109. The RP described the process by which processes are allocated an automatic or manual function, according to its own methodology, in turn taken from its ‘General Safety Requirements’ (Ref. 39, Ref. 40). The response to RQ-UKHPR1000-1124, includes details about the boron control functions performed by the boron control system, and can be separated into two types (Ref. 3):

- Type A: Operational functions, which are used in the slow water chemistry control during plant daily operation.
- Type B: Safety functions, which are used under an emergency situation.

110. Therefore, some boron functions involved in emergency situations are automatic but the majority are largely manual functions. The RP’s approach to the balance between manual and automatic functions has been assessed by my Human Factors colleague (Ref. 41).

111. The RP identifies two scenarios where the reduction in boron enrichment through the cycle introduces additional risks. As described above, the enrichment in the BAST gradually reduces during a cycle. However, the water in the BAST should also be capable of taking the core from power operation to cold shutdown. Although the RP has estimated that the enrichment in the BAST water will not fall below 35 at% in one cycle, the ^{10}B abundance in the BAST is monitored frequently, and if the abundance falls below 35 at%, boric acid is supplied from the REA [RBWMS].

112. The second risk arises due to the use of water from the IRWST during shutdown, and the RP suggests that over the course of several cycles, the abundance of ^{10}B in the IRWST may become depleted. To overcome this, boric acid of the target enrichment is injected directly into the primary circuit during shutdown, as well as fresh make-up using a much higher enrichment (up to 96 at%) into the BASTs, again via the REA [RBWMS]. The REA [RBWMS] is therefore required to make up boric acid solutions of differing enrichments. In response to a query about control of boron enrichment in the REA [RBWMS] in RQ-UKHPR1000-1124, the RP confirmed that the part of the REA [RBWMS] that controls boric acid mixing and distribution (the REA1) is normally operating at 35 to 39 at%, but can also be used to make up solutions of much higher enrichment; all other parts of the REA [RBWMS] only use 35 to 39 at% (Ref. 3). The response also identified that prior to transfer of the make-up water, samples are taken to measure the concentration and enrichment, to ensure the correct boric acid solution has been prepared. I also note that the boric acid solutions stored in the BASTs are sampled and monitored for the correct boron enrichment and concentration, prior to use in the RCP [RCS].

113. The capability of the sampling and monitoring system to support the control of boron enrichment is assessed in sub-section 4.2.8.

Boron Dilution

114. There are two types of boron dilution fault; homogeneous and heterogeneous:

- Homogeneous dilution occurs when the boron concentration gradually reduces until a concentration is reached where the control rods can no longer adequately control reactivity.
- Heterogeneous dilution occurs when a large amount of un-borated water enters the core, either from an internal or external source.

115. As boron dilution faults can affect reactivity, justification that these effects have been mitigated is a fundamental aspect of the safety case. As part of the RO that I raised for the RP to justify control of boron chemistry, I also raised an RO Action (ROA) – ROA2 – which required the RP to demonstrate that the risks associated with boron dilution faults have been reduced SFAIRP, as the safety case did not contain adequate justification at that point (Ref. 4).

116. The closure of ROA2 on boron dilution faults was led by colleagues from Fault Studies, and they considered both heterogeneous and homogeneous faults as part of their assessment of the RO closure (Ref. 42). I also reviewed aspects of the justification of homogeneous faults, as these are pertinent to the chemistry assessment, and this sub-section includes my consideration of the RP's safety case for this topic.

117. For the UK HPR1000, the method of detecting a boron dilution fault is the core monitoring system, and not the boron monitoring and sampling system. The fault schedule does not make any claims against the primary circuit monitoring and sampling system, and as such, a simple boron monitoring regime has been included within the generic design (as was described in the response to RQ-UKHPR1000-1124 (Ref. 3)). This is discussed further in sub-section 4.2.8. As well as boron concentration dilution, the use of EBA means that boron enrichment dilution is an additional fault that must be controlled.

118. The RP describes two homogeneous boron dilution initiating events:

- Low boron concentration in the reactor coolant due to malfunction of the RCV [CVCS], REA [RBWMS] and the TEP [CSTS]; and
- Boron dilution due to rupture of one heat exchanger tube.

119. The RP has identified that to minimise inadvertent human error, isolation of the RCV [CVCS] in the event of a dilution event will be automatic rather than being performed manually. I also note that the chemistry limits and conditions document includes a summary of the systems and their safety functional requirements for the control of boron concentration (Ref. 30).

120. During Step 4 of GDA, and in conjunction with my assessment of the work that the RP produced in order to close out the first action of RO-UKHPR1000-0031, I reviewed the safety case information for the control of boron, which includes prevention of dilution faults. I also commissioned a TSC to review the function of the systems that support the primary circuit, whose scope included the control of boron concentration.

121. As I described in sub-section 4.2.2.4, my TSC was generally satisfied with the arrangements that the RP has put in place to mitigate the risk of boron dilution events in normal operations for homogeneous faults. I have assessed how the RP intends to control the make-up sources of water for boron concentration and enrichment, and the methods it has developed to alert the operators to dilution events. Although my Fault Studies colleagues have led on the closure of RO-UKHPR1000-0031 Action 2, from the detailed assessment I have carried out, I am content that from a chemistry perspective, the RP has presented an adequate case to demonstrate that the risks of homogeneous boron dilution events have been reduced SFAIRP.

122. All of the matters raised as part of the closure of RO-UKHPR1000-0031 have been dealt with in this report; I raised one matter that was relevant to this assessment, which became a minor shortfall, and this is addressed in sub-section 4.2.8.

123. As I have discussed in sub-section 4.2.8, the primary circuit sampling and monitoring system includes provisions for boron analysis. Although the core monitoring system is the primary means of identifying boron dilution faults, the sampling and monitoring system is another mitigation in prevention of these faults, as it will forewarn operators

of errors in primary coolant make-up, as well as equipment faults. I will discuss the effectiveness of the system in sub-section 4.2.8, but for these reasons, it is important that the sampling system has been shown to be capable of delivering its function adequately.

4.2.2.5 Hydrogen Addition

124. Like all PWRs, the UK HPR1000 requires hydrogen to be dosed into the primary circuit, to help the suppression of oxidising species produced in the core, which helps to minimise corrosion of the primary circuit. For the UK HPR1000, the target hydrogen concentration in the primary circuit is between [redacted] and 50 $cm^3\ kg^{-1}$. Hydrogen is continuously lost from the primary circuit via a number of mechanisms:

- Radiolytic reactions
- Diffusion losses
- Degasification processes

125. Therefore, small concentrations of hydrogen are required to be dosed to the primary circuit during normal operations and during start-up. In most PWRs, hydrogen is added by pressurising the ullage of the VCT with hydrogen gas. The hydrogen concentration in the coolant is thus controlled by Henry's law, which is a simple and reliable method. However, if this were attempted in UK HPR1000, there would be between 5 and 15 m^3 of hydrogen gas in the VCT and the low flow through the VCT would need a proportionally higher hydrogen concentration in the cover gas, or higher operating pressure. Both of these factors would result in a significant risk of deflagration if the VCT leaked or if there was ingress of air. Instead, a hydrogenation station is included in the generic design. During Step 4 of GDA, I commissioned a TSC to assess the design of the hydrogenation station, and the following is a summary of their assessment (Ref. 34). Details of the design of the hydrogenation station can be found in PCSR Chapter 21, and the relevant SDM (Ref. 5, Ref. 35).

126. It is stated by the RP that more than 90% of the letdown flow rate is passed through the hydrogenation station, with the remaining 10% being routed to the VCT (Ref. 43). An assumed flow rate of 90% of the normal letdown flow rate has been used in the theoretical approach by the RP. If the letdown flow rate was increased to the hydrogenation station, then this would increase the demands upon the system. The RP states that increased flow is achievable from the use of two charging pumps, therefore the 90% letdown flow rate is reasonable (Ref. 43).

127. The use of a hydrogenation station compared to a traditional VCT over-pressure approach is claimed by the RP to offer the following benefits:

- Reduced hydrogen volume of 0.1 m^3 for a hydrogenation station compared to 8 m^3 of the traditional VCT approach. This reduces the risk of a hydrogen explosion and fire.
- Undissolved hydrogen can be re-used in the separator (if the hydrogen is still undissolved after the mixing pipe, the undissolved gas will be retained in the gas separator for re-use).
- The VCT no longer requires replacement of hydrogen with a nitrogen blanket, simplifying the design of the VCT. It should be noted that replacement of the hydrogen in the gas separator is still required at shutdown.

128. The RP identifies some drawbacks associated with this approach. The VCT will have a nitrogen blanket that will result in increased carbon-14 (^{14}C, from activation of dissolved nitrogen) and an increased ammonia (NH_3) concentration in the primary circuit fluid. The production of ^{14}C from this mechanism is considered in sub-section 4.2.3.3. In reply to RQ-UKHPR1000-1121, the RP indicated that ammonia concentration is likely to be not significantly higher than plants that use hydrogen as a

cover gas. The RP reports measured readings of less than [redacted] ppm in CPR1000 reactors (which use hydrogen as cover gas), and between [redacted] and [redacted] ppm in Taishan, which uses nitrogen (Ref. 3). I was satisfied with this response.

129. I raised RQ-UKHPR1000-1241 about the recirculation of gases from the gas separator to the water jet pump (Ref. 3). Radioactive gases in the aqueous phase will degas in the gas separator, resulting in an increased gas phase activity. The response highlighted that any gases would be sent for recycling to the Gaseous Waste Treatment System (TEG [GWTS]) and be replaced with fresh hydrogen (Ref. 34). I was satisfied with this response.

130. The approach presented by the RP within the safety case relies upon a theoretical hydrogen balance, supported by an experimental programme conducted by the RP (Ref. 43). The hydrogen balance as presented is appropriate and my TSC concluded that it has considered reasonable inputs and losses of hydrogen. Hydrogen is only added via the hydrogenation station and a justification is presented that the hydrogen pressure is sufficient to achieve a target concentration of 35 $cm^3 kg^{-1}$. I raised two RQs (RQ-UKHPR1000-1241 and RQ-UKHPR1000-1291) in this area in order to obtain confidence that the hydrogenation station can deliver the required hydrogen concentration (Ref. 3).

131. My TSC reviewed the losses of hydrogen from the system identified by the RP (Ref. 34). Of note were the assumed losses from diffusion through the SGs. The RP has used data from CPR1000 plants, but the UK HPR1000 uses thinner SG tubing; the RP has used a "realism factor" to account for this difference. Although a justification from first principles of hydrogen diffusion would have provided supporting evidence, the RP presented some limited evidence from the RP's operating plants, which supported the use of the "realism factor." I consider that for GDA this is reasonable, but the licensee should develop the evidence to support this claim, and I therefore consider this to be a minor shortfall.

132. The pressure within the gas separator vessel is controlled using the VCT pressure. Operating the VCT at a higher concentration therefore allows an increased hydrogen concentration. Using the theoretical model developed, the hydrogen concentration achievable is shown in Table 2:

Table 2: Volume Control Tank Hydrogen Concentration Achievable for a Given Pressure **(Ref. 43)**

Pressure (bara)	Hydrogen Concentration ($cm^3 kg^{-1}$)
2.9	25
3.3	30
3.7	35

133. The theoretical model is further supported by data from an experimental rig that simulated the hydrogenation station. The full basis and design of the rig was not available for assessment. However, a comparison against the CPR1000 design has also been assessed and used to validate the experimental setup. It is reassuring that an experimental rig has been constructed to demonstrate the performance of the hydrogenation station.

134. Only very limited OPEX from the RP or other vendors was presented to substantiate the design. The lack of operating data at 35 $cm^3 kg^{-1}$ from the RP is understandable due to Chinese plants not typically operating at 35 $cm^3 kg^{-1}$. The safety case states that

one unit in China with the hydrogenation station is able to maintain a target concentration of 1.5 – 2.0 ppm with one charging pump in service and 2.0 – 2.5 ppm with two charging pumps in service (equivalent to 17 – 28 $cm^3 kg^{-1}$). However, it is also claimed for the UK HPR1000 that the desired operating concentration can be achieved using one charging pump (Ref. 43). Some data from the experimental rig is presented that shows the UK HPR1000 can achieve a higher hydrogen concentration due to additional mixing elements in the mixing pipe of the hydrogenation station. This indicates that the performance is over twice that of the original design.

135. I judge that the RP has not provided sufficient evidence to fully justify the design of the hydrogenation station. I do have reasonable confidence that the UK HPR1000 hydrogenation station has the ability to achieve the hydrogen concentrations required in the primary circuit, due to my knowledge of other similar systems. I therefore consider this to be an Assessment Finding.

AF-UKHPR1000-0053 – The licensee shall, as part of detailed design, justify that the design and operation of the hydrogenation station reduces relevant risks so far as is reasonably practicable.

136. The ability of the hydrogenation station to achieve 35 $cm^3 kg^{-1}$ was assessed against the most demanding plant operations. Discussions with the RP identified that the greatest demand for hydrogen was at the end of a fuel cycle. This arises due to water movements for dilution and boration when non-hydrogenated water is injected into the circuit. At the end of a fuel cycle, the primary circuit is diluted to maintain a coordinated chemistry but prior to shutdown, the primary circuit is then borated, requiring a greater feedwater make-up rate. The RP has demonstrated that the design is able to cope with this situation and maintain the desired hydrogen concentration during foreseeable plant operations (Ref. 43).

137. My TSC considered both periods during which the letdown flow rate is increased (and leads to a reduction in hydrogen concentration), and during start-up where there is a requirement to achieve [redacted] $cm^3 kg^{-1}$ within 5.5 hours, and they were satisfied with the evidence provided by the RP (Ref. 34).

4.2.2.6 Other Species

138. The chemical additives of hydrazine, enriched lithium hydroxide (^7LiOH) and hydrogen peroxide are injected into the primary circuit by the chemical addition unit of the RCV [CVCS] (Ref. 33). My TSC reviewed the description of the addition systems and concluded that the technology was similar to that used in other PWRs (Ref. 34).

139. Most PWR plants have safety limits and action levels associated with the primary coolant pH. In order to limit corrosion within the primary circuit, lithium hydroxide is added. As the boron concentration changes through the cycle, so must the lithium concentration. Lithium is also produced during a cycle from the boron neutron absorption reaction. In the UK HPR1000, lithium is enriched in ^7Li, as ^6Li is a significant source of ^3H.

140. The arrangement for the addition and removal of ^7LiOH through the cation bed represents a standard approach used by many PWRs (Ref. 34). In response to RQ-UKHPR1000-1124, the RP stated that the chemical addition unit is not a safety classified system and therefore no redundancy is required (Ref. 3). The RP also stated that power raising or the commencement of power operation will not proceed if a chemical parameter cannot be met. The reasoning behind not having redundancy is that failure of the system would impact operability rather than nuclear safety. This arrangement is consistent with other PWR designs.

141. The design for the hydrazine and hydrogen peroxide addition system features one tank which is reported to prevent inadvertent addition to the wrong vessel. With the hydrazine and hydrogen peroxide additions occurring at different times, this arrangement appears appropriate provided operational safeguards are in place. The tank sizing appears appropriate based on the volume of chemicals to be injected and alignment with the reference plant, FCG3.

142. Whilst zinc addition is part of the proposed primary circuit chemistry, (assessed in section 4.2.6), the detailed design of the zinc injection plant is not part of GDA, though I am aware of many instances of zinc injection being employed successfully at power plants around the world. In response to RQ-UKHPR1000-0701, the RP also provided assurance that it was feasible to include zinc injection equipment within the reference design, in terms of plant compatibility and spatial availability (Ref. 3).

4.2.2.7 Strengths

143. Following the resolution of RO-UKHPR1000-0031, the RP has provided a coherent, detailed description of the boron control system, which clearly identifies key risks and hazards and how these have been mitigated, including an adequate description of homogeneous boron dilution faults.

144. The decision to use EBA at the enrichment level selected by the RP is clearly justified as the option which reduces risks SFAIRP and is supported by a detailed description of the optioneering process.

145. The safety case adequately justifies how chemical species will be added to the primary circuit, and justifies any risks have been reduced SFAIRP.

4.2.2.8 Outcomes

146. Based upon the assessment of the primary circuit chemical control systems in UK HPR1000 described in sub-section 4.2.2 above, I identified two Assessment Findings which need to be addressed by the licensee, which concern the inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011), and a justification that the design of the hydrogenation station reduces relevant risks SFAIRP (AF-UKHPR1000-0053).

147. I also identified three minor shortfalls as discussed in sub-section 4.2.2 above.

4.2.2.9 Conclusions

148. I am content that the RP has produced an adequate safety case to justify the operation and control of the RCV [CVCS] and associated systems such as the REA [RBWMS] and TEP [CSTS]. The RP has demonstrated that these systems can maintain the chosen primary circuit chemistry during normal operations. Although I have raised two Assessment Findings, I am content that the RP has justified these systems adequately for this stage of the development of the generic design of the UK HPR1000.

149. Based on the outcome of my assessment of the primary circuit chemical control systems in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

150. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects important to safety and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). SAP FA.2 provides guidance on the identification of faults. I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear

reactors, and chemistry assessment, and used NS-TAST-GD-006 which covers Design Basis Analysis (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the safety case.

4.2.3 Materials, Chemistry and Radioactivity

151. The radioactivity that is carried by the primary coolant is a principal source of Operator Radiation Exposure (ORE) and routine radioactive wastes in a PWR. The radioactivity also acts as a potential source term in accidents.

152. The main source of the ORE in a PWR (as much as 90%) can arise from activated CPs. Corrosion of the metallic items of the plant can cause CPs to enter the primary coolant, which can then be activated in the radiation field within the core. The major source of this is fuel deposits (this is assessed in detail in sub-section 4.2.5). Other sources of radioactivity arise from activation of the additives or impurities within the coolant, and releases of fission products from the fuel clad, either through diffusion or more directly from cladding defects.

153. There are two main factors which control the speciation and quantity of CPs. The first factor is the selection of materials that come into contact with the primary coolant. Essentially, all of the activated CPs within a PWR originate from the release of material into the coolant, either via corrosion or wear of the primary circuit materials in contact with the coolant. The second factor is the choice of the chemistry regime of the primary circuit; this has a significant impact on the quantities and distribution of CPs about the circuit.

154. Even in the best controlled plants, general corrosion (even if at a low rate) results in the transport of non-radioactive cobalt and nickel through the core to be activated to the intensely radioactive isotopes cobalt-58 (^{58}Co) and cobalt-59 (^{60}Co), which are the main sources of ORE. There are four main methods of minimising the amount of radiation via activated CP:

- Choosing materials that minimise the release of elements such as cobalt and nickel into the primary coolant.
- Adopting a primary circuit chemistry regime with the aim to reduce corrosion.
- Using the RCV [CVCS] to try and remove as many of the precursors and radioactive products as possible.
- Maintain high purity make-up sources and avoid/minimise deleterious impurities.

155. The topic of how material choices and chemistry affect the generation and transport of radioactivity is broad. To assess this part of the generic safety case, I selected a broad assessment sample, to be able to judge whether the RP had adequately justified that radioactivity had been reduced SFAIRP.

156. At the beginning of Step 4 of GDA, Chapter 21 of the PCSR identified many of the materials choices made in the design of the primary circuit, and the primary circuit chemistry regime which was to be adopted (Ref. 5). However, there was no coherent or adequate justification of how radioactivity had been minimised and that the associated risks had been reduced SFAIRP, and I therefore raised RO-UKHPR1000-0026. The RP's response to this RO was to produce a single deliverable, a 'Minimisation of Radioactivity Route Map Report', which I assess in the following sections (Ref. 44).

157. A further important aspect of the justification that radioactivity has been reduced SFAIRP is an analysis of the generation and transport of ^{3}H within the primary circuit. ^{3}H is produced constantly within the primary circuit and there are only limited mitigation options available. During Step 4 of GDA, the information underpinning the RP's

estimate for ^{3}H production in UK HPR1000 and the associated limits and conditions necessary in the interests of safety, were based on OPEX from Chinese PWRs which have different design features and are operated in a different way, to the intended operations for UK HPR1000. I therefore raised RO-UKHPR1000-0049 (Ref. 4). I will assess the response to this RO in the following sections.

158. Several other areas of the assessment are relevant to radioactivity:

- Zinc (as zinc acetate) will be added to the coolant to further reduce out-of-core fields, assessed in sub-section 4.2.6.
- Hydrogen will be added to the coolant to prevent oxidising conditions, assessed in sub-section 4.2.7.
- The fission products and actinides generated from tramp uranium, or which escape from defective fuel elements, are captured by the waste-treatment systems, described in sub-section 4.4.3.
- My assessment of the radioactive source term (mainly iodine) during accidents is presented in sub-section 4.5.

159. As part of my assessment, I used the guidance provided in the SAPs ECH.1, ECH.3, EHT.5 and EMC.16, in addition to NS-TAST-GD-088 and NS-TAST-GD-089 (Ref. 2, Ref. 6).

4.2.3.1 Material Choices

160. From a Chemistry perspective the CPs released from materials which are in contact with the primary coolant are an important characteristic of the primary circuit systems. CPs released from metallic corrosion or wear mechanisms, are transported into the core with the coolant flow, where they may become activated in the radiation field. The selection of materials in contact with the coolant is consequently a primary factor that determines the susceptibility of the reactor to the production of activated CPs. The chemistry has a direct influence on the extent of general corrosion; in addition it can influence the redistribution of released material around the RCP [RCS] and connected systems, depending on factors such as solubility or speciation. The significant nuclides produced from the UK HPR1000 RCP [RCS] materials are typical of PWRs, the main radionuclides that contribute to ORE are ^{58}Co and ^{60}Co.

161. There are three main sources of activated cobalt that arise from CPs:

- Components made from cobalt-based alloy
- Steels and alloys which contain traces of cobalt
- Nickel-based alloys in bulk materials

162. The activated CP is a relatively small proportion of the overall CPs formed within the primary circuit. However, several thousand square metres of material are in contact with the primary circuit and the associated (albeit relatively small) release of material can give rise to quantities of nuclides which are significant for radiological protection. The surface area of each alloy is therefore an important factor. Alloys containing trace cobalt with a small surface area should make little contribution to radioactivity build-up, whereas those with large surface areas like the SG tubes, will have a bigger effect. Cobalt-based alloys, similarly, will have a relatively large effect despite having a comparatively low surface area. From the data provided by the RP in Step 4 of GDA, I have deduced the total wetted surface areas for the different groups of alloys to be as shown in Table 3.

Table 3: Assumed Wetted Surface Areas of the Alloy Groups of SSCs in the UK HPR1000 Primary Circuit

Alloy Group	Approximate Total Wetted Surface Area (m^2)	Typical Locations
Nickel-based Alloy	25,700	SG tubes & divider plate, RPV, Reactor Vessel Internals (RVI) Cladding of RPV, RVI, SG tubesheet
Stainless Steel	2,300(1)	RCP, MCL, cladding of SG, RPV, RVI, PZR
High Cobalt Alloy	5(2)	Control Rod Drive Mechanisms (CRDM), RVI & Valves

(1) Noted to be 170 m^2 in (Ref. 45)
(2) Includes a contribution from the RCP in (Ref. 45)

163. General corrosion is greatest at the beginning of a reactor's life, and whilst commissioning processes will help to passivate the alloy surfaces, a small amount of general corrosion will occur at initial start-up, but this should reduce over the initial cycles of operation as the alloy surfaces continue to passivate. I consider that the target should be to achieve a high factor of improvement in the least number of cycles, the need for careful chemistry control being a vital part of this process.

164. Cobalt-based alloys (principally Stellites™) are hard wearing alloys and, consequently, are commonly used in PWR components such as CRDMs, valve seats and wear pads. Cobalt input into the primary circuit from Stellite™ arises from four potential mechanisms, namely, wear, corrosion, erosion-corrosion (where large pressure drops across valves induce large flow velocities) and valve lapping. It has been demonstrated that these alloys have contributed much of the cobalt which caused significant radioactivity in older PWRs (Ref. 46, Ref. 47). It is not certain precisely how much of this was due to operational wear, corrosion or poor maintenance (for instance, if allowed, in-situ valve lapping may release cobalt particulate into the circuit).

165. As Stellite™ is progressively removed from the design, trace cobalt levels in other steels used to construct the reactor have an increasing contribution to the relative proportion of ^{60}Co. High levels of cobalt impurity in structural metals are associated with higher ^{60}Co levels arising from general corrosion of the structural materials. Even small changes in the trace cobalt concentration of these materials can be significant given the large surface areas (Table 3) that are in contact with the coolant.

166. Associated with this topic, the RP makes claim SC21.3 in Chapter 21 of the PCSR for the chemistry aspects of the material selection to minimise the presence of cobalt SFAIRP (Ref. 5):

- 3.3.10.SC21.3: Radioactivity level in the primary circuit is reduced SFAIRP to minimise worker and public dose.

167. The main reports detailed in the PCSR of relevance to my assessment of cobalt within the primary circuit materials are: 'Material Selection Summary Report', 'ALARP Assessments for Applications of Cobalt Based Alloy in Reactor Coolant Pump', 'Topic Report on Application of Cobalt Based Alloy in SSCs'; 'Application Analysis of Cobalt Based Alloy in Valves' (Ref. 48, Ref. 49, Ref. 50, Ref. 51).

168. The principal SAPs relevant to my assessment of cobalt and radioactivity are EAD.1 to EAD.2, ECH.1, ECH.2 and EMC.13. I also raised three RQs, RQ-UKHPR1000-1132, RQ-UKHPR1000-1390 and RQ-UKHPR1000-1633 to gain further clarification and evidence to support the claim made in the PCSR (Ref. 3).

169. As part of my assessment, I requested further information on the RP's analyses of the apportionment of ^{60}Co sources between key SSCs within UK HPR1000. These analyses form part of the evidence to underpin its claim that risks are reduced SFAIRP which will be discussed in the following sub-sections. A number of contradictions were present within the safety case and the responses provided did not create a clear picture of what the relative contributions in the UK HPR1000 were. During my assessment, the apportionment of ^{60}Co between SSCs changed dramatically, largely based on an Electric Power Research Institute (EPRI) reference, without any consideration of its applicability to the generic UK HPR1000 design. In summary:

- Initial estimates presented by the RP for ^{59}Co release and contribution of ^{60}Co were based on "empirical judgement of experienced experts" of CPR1000 OPEX. As a result, this did not include any optimisation for Stellite^{TM} or trace cobalt levels within the bulk materials. Similar contributions were given for the CRDM, RVI, pumps (RCP and RCV) with the bulk materials contributing 5%.
- The RP stated that data from the EDF M 310 reactor (upon which CPR1000 was developed, with similar application of Stellite^{TM}) on ^{60}Co change following component replacements support this judgement. The supporting EDF data was not made available as part of my assessment. No such CPR1000 data was available, as, to date, no such replacements have been made.
- Using only corrosion release rates provided by the RP, I obtained a similar apportionment of cobalt, which suggested the impact of wear had not been taken into account. In response to RQ-UKHPR1000-1639, where I raised this, the RP provided an updated assessment based on this single Stellite^{TM} corrosion release rate, and SSC specific wear rates taken from a supporting report (Ref. 3, Ref. 52). Revised estimates showed the valves to contribute >90% of the ^{60}Co before removal of Stellite^{TM}. Following optimisation of Stellite^{TM} in valves (as detailed below), the CRDMs were estimated to contribute to ~80%, SG tubes at ~10%, RVI <10% and valves <1%.

170. The RP maintains that the initial estimates of the ^{60}Co apportionment amongst the SSCs is "reasonable". However, it took the revised assessment forward as a demonstration that risks due to Stellite^{TM} are reduced SFAIRP. As the revised estimates are based on US plant data from 1984, I do not consider that the applicability of this analysis to UK HPR1000 has been demonstrated. The position claimed by the RP for the use of Stellite^{TM} in valves (discussed in the following paragraphs) relies on this US plant data and depends on a robust demonstration of the apportionment of ^{60}Co between the different valve groups. The two different methods for determining which SSCs contribute most to ^{60}Co levels give very different results, and do not take into account other available OPEX in this area. I do not consider that the RP has provided an adequate understanding of the impact of the different SSCs to the ^{60}Co in UK HPR1000. I consider this to be an Assessment Finding.

AF-UKHPR1000-0112 – The licensee shall, as part of detailed design, justify that the cobalt inventory within UK HPR1000 has been optimised, taking account of the relevant risks. The justification should include the impacts of different operating conditions and material choices on radioactivity, worker doses and component reliability.

Cobalt-based Alloy

171. In the UK HPR1000, Stellite^{TM} has been eliminated from the RCP. For the reference plant (FCG3), Stellite^{TM} has also been removed from RCV [CVCS] pumps. The RCV [CVCS] pump was not originally within the RP's GDA scope, and so a detailed justification has not been made for the material replacement. I consider this to be part of Assessment Finding AF-UKHPR1000-0112. The replacement material, a martensitic stainless steel, forms a sleeve over the pump shaft at the graphite bearing. The RP refers to a number of plants (such as Angra 2/3, Changjiang 1, Gösgen, Isar 2, Qinshan and Obrigheim) as OPEX to support this material selection. In comparison to Stellite^{TM}, where the radial bearing would require inspection and replacement every 12 years, the RP notes a significantly reduced maintenance cycle with the replacement material requiring inspection every 4 years and replacement every 10 years. Nevertheless, the RP claim an estimated reduction to ^{60}Co and ORE of 3% and 1% respectively for the RCP and a reduction to ^{59}Co of 20% for the RCP and RCV [CVCS] pumps combined (Ref. 49, Ref. 45). Whilst I accept an increased maintenance schedule will lessen the benefit to ORE, the precise impact on ^{60}Co is not clear. This is part of Assessment Finding AF-UKHPR1000-0112.

172. In the UK HPR1000, Stellite^{TM} is used within the following components in the RCP [RCS]:

- CRDM, 3.1 m^2
- RVI, 0.561 m^2
- A number of valves, 0.50 - 2.72 m^2 (dependent on final selection by the licensee)

173. The RP claim that the Stellite^{TM} within the CRDM and RVI are in line with that of the CPR1000 reactors, which are optimised in terms of Stellite^{TM} use. The RP's assessments scored alternative alloys lower than Stellite^{TM} on the basis of reliability, technical maturity, RGP, ORE for additional maintenance as well as the impact any changes would have on cost and schedule arising from the requirement for substantial design changes and testing that would be required. Overall, I am content that this is reasonable. However, as this is an on-going area of research, the licensee for UK HPR1000 should ensure no suitable alternative to Stellite^{TM} exists before acquiring significant equipment containing hard-facings in contact with primary coolant.

174. For the valves, the approach taken by the RP is to provide a process, by which decisions over Stellite^{TM} use can be made by the licensee. Of the 3905 valves which are part of, or connected to, the main primary circuit, only 292 are taken forward for consideration of alternative hard facing materials, on the basis that the other valves' medium does not flow into the core.

175. The process specifies the use of Stellite^{TM} in valves according to three categories: those for which there is no limit on Stellite^{TM} use, those where its use is prohibited and those where the licensee is free to choose, although its use is not recommended. The factors considered are whether the medium exiting the valve will flow into the core, valve size, the frequency of use and sealing requirements, and placement of the valve (upstream or downstream) relative to the filter and demineralisers within the RCV [CVCS] or TEP [CSTS]. The outcome of the process is summarised in Table 4.

Table 4: Valve Categories and Wetted Surface Area **(Ref. 51)**

Stellite™ use Category	Number of Valves	Position relative to filter and demineraliser	Total Wetted Surface Area (m^2)	Contribution to ^{60}Co (%)
Prohibited	54	Downstream	0.77	99
licensee choice - not recommended	113	Downstream	1.30	0.6
		Upstream	0.15	0
No limit	125	Downstream	-	
		Upstream	0.50	0

176. The basis of the process is that valves which contribute the most to ^{60}Co will use alternative alloys, whilst those that have only a small impact may retain Stellite™. The analysis provided by the RP indicates that the valves in the final "no limits" category contribute <1% of the total ^{60}Co valve contribution before applying the proposed hard facing material process. As noted above, these calculations are based on a single corrosion release rate for each component, and wear rates taken from a 1984 EPRI publication based on plant data of that time, and as such, the applicability of this analysis to UK HPR1000 has not been demonstrated (Ref. 52).

177. Application of this process would result in valves for which an alternative material could easily meet the functional requirements retaining Stellite™; this is in contrast to RGP, see for example the guidance provided by EPRI (Ref. 53). Whilst in principle, the selection criteria are tailored for reduction of ^{60}Co, I am not yet convinced that the process reflects a position in which risks have been reduced SFAIRP as it does not fully take into account RGP, technical feasibility of alternative materials or the hierarchy of control measures. I would expect the licensee to further justify which valves could be subject to Stellite™ replacement. I consider this to be an Assessment Finding.

AF-UKHPR1000-0113 – The licensee shall, as part of detailed design, justify which valves will include cobalt-based hard facings. This should focus on those which do not fall into the "prohibited" category according to procurement specifications. The justification should demonstrate that relevant risks have been reduced so far as is reasonably practicable.

178. Overall, the RP has made reasonable arguments to support the claim that the high cobalt alloy inventory in these components of the UK HPR1000 has been reduced SFAIRP, based upon the current knowledge in this field, except for my questions regarding the retention in valves, for which I have still to see good justification. Currently some use of cobalt alloys seems unavoidable and other chemistry measures outlined in this report can be optimised to make control of ^{60}Co releases. However, as this is an on-going area of research, the licensee for UK HPR1000 should ensure no suitable alternative to Stellite™ exists before acquiring significant equipment containing hard-facings in contact with primary coolant. It is important that the licensee ensures that there are sufficient instructions for fabricators and operators to install, commission and maintain any hard-facing materials that may give rise to ^{60}Co dose.

179. It is also important when considering any Stellite™ replacement that the alternative is not actually worse in terms of either its mechanical properties or the corrosion of other elements in the alloy. Use of alternative, replacement alloys may result in different amounts of wear and corrosion. The RP's consideration of these aspects of the use of

Stellite™ and their low cobalt replacements were outside the scope of my Chemistry assessment, however.

Trace Cobalt Levels

180. It is not possible to completely eliminate cobalt from the bulk materials in UK HPR1000. In order to reduce the ^{60}Co which may arise from corrosion of these materials, the RP has specified controls for the trace cobalt concentrations of the SSCs in contact with the reactor coolant as summarised in Table 5 (Ref. 48). These specified levels are claimed to be at least consistent with the requirements of ‘Design and Construction Rules for Mechanical Components of PWR Nuclear Islands (RCC-M)’, previous designs developed and built by CGN, and other PWR plants worldwide (Ref. 54).

Table 5: Maximum Cobalt Concentrations in the Bulk Materials of SSCs in UK HPR1000 **(Ref. 48)**

SSC		Material group	Maximum Specified Cobalt Content (%)
General	Various	Stainless steels	0.1
		Stainless steel filler metal (clad & welds)	0.06
		Ni-based alloys	0.06
		Ni-based alloy filler (clad & welds)	0.06
SSC Specific	SG tubes	Ni-based alloy	0.015
	RVI	Stainless steel	0.06

181. Tighter controls on the trace cobalt levels are specified for the RVI and SG tubes to reflect the significance of these SSCs to ^{60}Co levels. The RVI are subject to high levels of neutron irradiation, activation can occur in-situ before release of CPs in addition to activation following CP deposition on the fuel clad surface; a higher relative proportion of the CPs from the RVI will therefore be activated than in lower neutron flux areas. The SG tubes represent over 80% of the primary surface in contact with the coolant, and, as a result, will transfer a relatively large quantity of CPs to the reactor core.

182. Within ‘Material Selection Report of Steam Generators’, the RP references the material specification from the reference plant FCG3, for “more details about the technical requirements of SG tubes”, which noted the SG tube material to have an average cobalt content ≤0.014%, maximum 0.016% (Ref. 55, Ref. 56). This specification follows a similar form to that of RCC-M, which specifies a maximum cobalt content in any one heat (the material from one molten batch) of 0.035%, with an average of cobalt content of the entire bundle to be ≤0.018%. It is not clear if the maximum 0.015% cobalt claimed in the PSCR reflects an increased or decreased cobalt content relative to the reference plant (Ref. 5). Similarly, the 0.06% cobalt specified in the stainless steel cladding and weld is claimed to be 0.05% cobalt in the PCSR (Ref. 48, Ref. 5). Whilst a minor point, these inconsistencies should be corrected by the licensee. This has been captured as part of AF-UKHPR1000-0112.

183. I am largely satisfied that the trace cobalt specifications provided by the RP are appropriate at the present stage, noting the small inconsistencies presented in some of the document submissions. The control and enforcement of such limits during manufacturing are out of scope for GDA, however, the RP has provided sample

documentation used for the reference plant FCG3 which adequately demonstrates how this might be achieved for UK HPR1000.

Bulk Materials

184. In addition to ^{60}Co and ^{58}Co, a number of other radioisotopes are produced by activation of CPs from the structural steel materials, including chromium-51 (^{51}Cr), iron-59 (^{59}Fe), manganese-54 (^{54}Mn) and nickel-63 (^{63}Ni). These materials are required to preserve the integrity of the pressure boundary and cannot be replaced. Instead, emphasis must be based on controlling the corrosion rate of these materials by chemistry optimisation and material treatments. Reductions in ^{60}Co will leave proportionally more ^{58}Co, as reduction in the quantity of nickel in contact with the coolant is more difficult in reactors.

185. Fuel design, manufacturing techniques and fuel management are outside the scope of my assessment. However, my assessment of the chemistry aspects of fuel clad integrity, which is relevant to this sub-argument is detailed in sub-section 4.2.5.1.

186. The main reports detailed in the PCSR of relevance to my assessment of the bulk materials selection are the ‘Material Selection Summary Report’, ‘Ageing and Degradation Justification Summary Report’ and ‘Topic report on Surface Treatment of SSCs’ (Ref. 48, Ref. 57, Ref. 58). I have also used ‘Material Selection Report of Steam Generator’ and ‘Ageing and Degradation of SG (Ref. 56, Ref. 59)’.

187. The principal SAPs relevant to my assessment of the bulk materials selection and radioactivity are ECH.1, ECH.2 and EHT.5. I also raised two RQs, RQ-UKHPR1000-1318 and RQ-UKHPR1000-1633 to gain further clarification and evidence to support the claim made in the PCSR (Ref. 3).

188. The RP has not presented a full, detailed justification for the materials of each individual SSC, but has provided a “grouping of SSCs” based on the service conditions (temperature, pressure, flow, chemical environment, safety classification and so on) and detailed material selection of representative SSCs (Ref. 48). I consider this to be proportionate for GDA, and any gaps in information will be addressed as part of normal business during the licencing phase. The RP has specified stainless steel and nickel-based alloys as the materials which will be in contact with the primary coolant (for full details, see sub-section 4.2.4.1). The grades of alloys selected have strict limits on trace elements such as sulphur and phosphorus, which impacts corrosion (amongst other material properties). These materials show good resistance to general corrosion and erosion by the primary coolant and selection of these materials can be considered normal practice for PWRs.

189. The SG tubing comprises over 80% of the total wetted surface area in the primary circuit, and only slight corrosion of such a large area will result in transfer of some CPs to the reactor core. As such, I chose to sample the SG tube material selection during my Step 4 assessment.

190. The RP has selected Alloy 690 Thermally Treated (TT) for the SG tubing in UK HPR1000. Alloy 690 is a nickel-based alloy with nickel content of ≥58.0% and TT refers to an additional thermal treatment after the final mill anneal stage during manufacture. Modern PWR SG tubing is typically composed of either Alloy 690 or Alloy 800. Alloy 800 is not a nickel-based alloy, with a nickel content of 30.0-35.0%. However, Alloy 800 was not selected for UK HPR1000 because it has a lower thermal conductivity and experimental studies indicate a higher susceptibility to Stress Corrosion Cracking (SCC), denting and foul corrosion (Ref. 60). I consider that the use of either material is RGP, and therefore I consider the selection of Alloy 690TT to be acceptable. Radiation fields in modern PWRs are low, partly as a result of the use of Alloy 690, however material selection is not the whole story. Manufacture and

commissioning processes can impact the passivation of SSCs' surfaces and hence the radiation fields arising from activated CPs. My assessment of the commissioning process for UK HPR1000 is detailed in sub-section 4.2.11.

Manufacturing, Surface Finishing and Surface Cleanliness

191. Most surface treatments applied to alloys within PWRs are intended to modify stresses or help with non-destructive testing, which is out of scope of this report. However, surface treatment also has an effect on both corrosion rates and radioactivity pickup.

192. The surface cleanliness is also important to reduce corrosion. Foreign material may be introduced as a result of operations such as manufacturing, installation, repair, maintenance, inspections, or other intrusive procedures. Similarly, specifications to minimise contamination of stainless steel and nickel-based alloys (such as by copper, low melting temperature alloys, mercury, and lead) should be included to minimise corrosion. Surface conditioning and cleanliness criteria for the commissioning phases are assessed in sub-section 4.2.11.

193. The RP has produced a topic report concerning the surface finishing of the UK HPR1000 SSCs (Ref. 58). The RP's selection of key SSCs for surface treatment was based on consideration of those which represent typical materials and large wetted surface areas, together with worldwide OPEX.

194. I requested in RQ-UKHPR1000-1318 a ranking of SSCs according to corrosion release and ORE to assess the adequacy of the RP's SSCs selection for surface treatment (Ref. 3). The response noted that the key contributor to CP release (resulting in ^{58}Co following activation in the core) was the SGs, whilst in terms of ORE, key contributors were maintenance activities involving valves, the RPV, SGs and RCPs. As the components which were the main contributor to ORE (valves) were not included in the initial SSCs selection, I requested further information in RQ-UKHPR1000-1633 on the surface treatments which would be applied to the key components which affected ORE from activated CP deposition (Ref. 3). The RP's response was that the manufacturing specifications for valves and RCP would be completed in site-specific stages, however the expected surface finishes would be as follows:

- Valves in contact with primary coolant Ra ≤ 6.3 µm
- RCP for ground and machined surfaces, with the exception of the pump shaft chromium carbide (Cr_3C_2) coating (Ra ≤ 0.6 µm) and the SiC30 seal ring (Ra ≤ 0.1 µm)
- Reactor and Fuel Pool walls (Ra ≤ 3.2 µm)

195. In addition, the RP provided details of specified surface finishes for the auxiliary systems (such as the RCV [CVCS]), which may impact the CP release. The RP stated that consideration had been given to tighter controls, but this would result in prohibitive cost increase, and that the final position reduced risks SFAIRP. I consider this to be acceptable for GDA, however, this should be reviewed and justified by the licensee on manufacturer selection during site-specific stages. I consider this to be a minor shortfall.

196. The RP includes reference to an additional pickling and passivation surface treatment for the MCL and RVI. In response to RQ-UKHPR1000-1633, the RP clarified that although this technique is included in the manufacturing specification of the MCL, it is not a mandatory requirement, and a manufacturer may choose to apply a different technique such as bright heat treatment to obtain an equivalent surface if approved by the designer in site-specific stages (Ref. 3).

197. The RP has not specified the manufacturers that would be used for the different SSCs for UK HPR1000 at the GDA phase, and detailed specifications will need to be

produced during site-specific stages. However, I requested information of other aspects of manufacture such as heat treatment and methods for residual stress relief which may have an impact on the CP release and deposition by modification to surface roughness and impact on microstructure (Ref. 61, Ref. 62). The RP responded to each request (RQ-UKHPR1000-1318 and RQ-UKHPR1000-1633 (Ref. 3)) with the surface roughness criteria only, which suggested it believed this to be the only important consideration. Manufacturing methods are generally in line with RGP; however this is an important omission to the generic UK HPR1000 safety case. I consider this to be an Assessment Finding.

AF-UKHPR1000-0114 – The licensee shall, as part of detailed design, demonstrate that the manufacturing routes of key structures systems and components reduce radioactivity so far as is reasonably practicable.

198. As part of my assessment, I sampled the SG tube manufacturing, as this can have a significant impact on ^{58}Co. In addition to the material specification and ageing and degradation reports (Ref. 55, Ref. 59), the RP provided a materials specification used for FCG3 SG tubing ‘Material Specification for Steam Generator Tubing Nickel-Chromium-Iron’ (Alloy 690) (Ref. 56). Together, these documents provide tube manufacturing and conditioning processes, including criteria for control of surface cleanliness during manufacture. The specifications and controls presented appear robust and consistent with RGP.

Other Materials – Antimony and Silver

199. Both antimony and silver components have been used in PWRs historically. Activation in the core of antimony and silver released by components will produce antimony-122 (^{122}Sb), antimony-124 (^{124}Sb), and antimony-125 (^{125}Sb), and silver-110m (^{110m}Ag) respectively. These nuclides are intensely radioactive and tend to plate out on surfaces outside the core and in fuel pools if released. Once plated out, they are difficult to remove.

200. The main report detailed in the PCSR of relevance to my assessment of cobalt within the primary circuit materials is ‘Material Selection Summary Report’ (Ref. 48). I also sampled ‘The Corrosion Product Source Term Analysis with UK HPR1000 Specific Design’, ‘SCCA - Description, Functional Requirements and Material Properties’ and ‘HARMONI RCCA - Description, Functional Requirements and Material Properties’ as part of my assessment (Ref. 45, Ref. 63, Ref. 64).

201. In the UK HPR1000, antimony in the form of antimony-beryllium pellets clad with stainless steel is used within the SNS rod. Whilst the rods are not expected to be exposed to the primary coolant, any failure of the clad material will result in dissolution and transfer of antimony to the coolant. The RP claim an improvement to the SNS clad has been made relative to older CGN units through the application of a “wear remedy” by an ion nitriding process to the clad, although this is not contained within the supporting Framatome document (Ref. 45, Ref. 63). I consider the use of antimony within the SNS rods to be an acceptable position for GDA.

202. In the UK HPR1000, silver is used in the seal gaskets of the RPV and is incorporated in the absorber rods of the RCCA. This is an improvement to the reference plant (FCG3), where silver is also retained in some seal gaskets in nuclear auxiliary systems.

203. The RPV seal gasket ensures sealing between the closure head and vessel assemblies and, in the UK HPR1000, takes the form of two C-sealing rings seated in grooves on the lower surface of the flange head. The RP retains silver on the outer surface of this component due to its good sealing performance, and claims there are no alternative materials which meet the functional requirements. It also maintains that

with a coolant contact area of <0.02 m^2 it is a low contributor to activated CPs and ORE; I consider this to be an acceptable position for GDA.

204. The control rods used for adjustment of reactor power in the UK HPR1000 contain a silver-based alloy (silver-indium-cadmium alloy) inside a stainless steel tube. The clad outer surface is ion nitrided to improve wear resistance. The RP also claim that an increase to the clad thickness has been implemented as a design improvement relative to older CGN units, although this is not contained within the supporting Framatome document (Ref. 45, Ref. 64). Like all PWR control rods these will have a defined service life and potentially suffer from swelling. Any silver that escaped from an aged or damaged control rod would present a significant contamination risk in a reactor. The management of control rods to ensure replacement before material is released should be part of the normal licensee activities.

205. Whilst I consider the material choices are acceptable for the GDA stage, specifications should be kept under review by the licensee and alternatives considered, if a substitution can safely be made, as part of normal business in the site-specific stages. Preventative surveillance of the SNS and RCCA rods should be reviewed and justified by the licensee, which I consider to be part of normal business in the site-specific stages.

4.2.3.2 Source Terms

206. Control of personnel doses and the minimisation of radiological wastes is an important objective for an operator of a reactor. The RP has therefore produced a suite of documents that, taken together, form the normal operation source term (Ref. 65, Ref. 66, Ref. 67). This can be defined as the types, quantities and physical and chemical forms of the radionuclides present in a nuclear facility that have the potential to give rise to exposure to radiation, radioactive waste or discharges to the environment.

207. The source term attempts to quantify the generation of radionuclides in the core region, and is classified into four groups:

- Fission products
- CPs
- Activation products
- Actinides

208. Most of the radionuclides are retained in the core region, notably by the fuel cladding. However, a proportion of them will either be produced in the primary coolant or pass through fuel cladding defects and enter the primary coolant. Once the radionuclides have entered the primary coolant, they can be transported to the various connected systems through various means, and a small proportion will ultimately be discharged into the environment. Assessment of the normal operation source term therefore requires a multidisciplinary approach, and colleagues from Radiological Protection, Nuclear Liabilities Regulation, and the Environment Agency have performed assessments in this area, although Chemistry took the lead of the overall assessment (Ref. 36, Ref. 68, Ref. 69).

209. To generate the source term, the RP has mostly used OPEX from the existing fleet of reactors in China that are operated by CGN. This data is therefore taken from plants that are similar, but not identical to the design of the UK HPR1000. Where OPEX is not available, the source term is calculated using theoretical calculations based on simple production and decay. Two values are calculated for the source term (Ref. 65):

- Realistic values – representing a best estimate of the activity concentration of radionuclides expected in the UK HPR1000 during normal operation.

- Design values – representing a conservative maximum for the source terms which can be considered a bounding limit for the plant design. It is expected that this limit would not be exceeded during operation and includes expected events which are expected to occur in the lifetime of the plant.

210. I asked a colleague from Radiological Protection to assess aspects of the source term, which focussed upon the choice of radionuclides, the use of OPEX and calculations, and the magnitude of the source term that was generated (Ref. 70). Comparisons were made between the source term for the UK HPR1000 and other similar designs, and it was concluded that the radionuclide activity concentrations calculated for the UK HPR1000 source term are reasonable. However, the review identified a number of shortfalls in terms of how OPEX had been used by the RP to develop the UK HPR1000 source term.

211. Firstly, the assessment found that not all of the OPEX referenced by the RP contained the expected level of detail in terms of the breadth of plant data supplied. This issue is part of Assessment Finding AF-UKHPR1000-0011, and aspects of the source term review findings have been included in this Assessment Finding. Secondly, the assessment found that the RP did not provide a systematic method for the selection of OPEX that forms the datasets used in the source term calculations. This is also part of Assessment Finding AF-UKHPR1000-0011, which also found that it was not always clear how the RP had selected the plant data, and where it was presented in the safety case (Ref. 70).

212. ONR’s assessment of the source term notes that zinc-65 (^{65}Zn) has not been included (Ref. 70). The RP has justified its omission on the basis that as Depleted Zinc Acetate (DZA, depleted in ^{64}Zn to minimise the production of ^{65}Zn) is used at a low concentration, the quantities are likely to be low, although no evidence is presented to support this judgement (Ref. 71). I have discussed the significance of this radionuclide with colleagues from Radiological Protection, Nuclear Liabilities Regulation and the Environment Agency, and I am content that ^{65}Zn is not the most significant of radionuclides in terms of operator dose or waste arisings (Ref. 72, Ref. 73). However, the licensee should consider whether to include ^{65}Zn in the source term, and therefore I consider this to be a minor shortfall.

213. Alongside the assessment by Radiological Protection, I raised four RQs (RQ-UKHPR1000-0690 to RQ-UKHPR1000-0693) to clarify various aspects of the source term from a Chemistry perspective, focussing on the primary circuit source term (Ref. 3). I was generally content with the approach taken, and the choice of radionuclides.

4.2.3.3 Minimisation of Radioactivity

214. Whilst the source term (sub-section 4.2.3.2) provides a best estimate of the generation and transport of the various radionuclides throughout the plant, its purpose is not to demonstrate that radioactivity has been reduced SFAIRP. During Step 3 of GDA, I asked the RP where this would be described and justified in the generic safety case. The RP argued that the source term provided a suitable demonstration, and that no further justification was required. To ensure adequate closure of this gap in the case, I raised RO-UKHPR1000-0026, which required the RP to complete four actions (Ref. 4):

- Produce a safety case route map for radioactivity in the generic UK HPR1000 design.
- Demonstrate that radioactivity in UK HPR1000 has been reduced SFAIRP.
- Demonstrate that the generic UK HPR1000 design is capable of minimising radioactivity in the primary circuit and connected systems.
- Identify any controls necessary to ensure radioactivity in UK HPR1000 will be minimised.

215. In response to RO-UKHPR1000-0026, the RP produced a single deliverable, a Route Map that would perform two functions (Ref. 44). Firstly, it would provide links to where more detailed evidence is contained within the safety case, and secondly, to provide an overarching demonstration that radioactivity in the UK HPR1000 has been reduced SFAIRP. As the Route Map attempts to describe all methods of reducing and controlling radioactivity, several other disciplines used the document as part of their assessment. The following sections cover those areas of assessment from a Chemistry perspective; the closure of this RO was coordinated with specialists from Radiological Protection, Nuclear Liabilities and Decommissioning, and the Environment Agency (Ref. 74).

216. The Route Map itself consists of a table that summarises the arguments and evidence that seek to demonstrate that radioactivity has been reduced SFAIRP, as well as references where that evidence can be found (Ref. 44). A full list of relevant references is provided, alongside details of which technical area produced the documents. Two main arguments are presented:

- Generation of radionuclides has been minimised
- Radionuclide transport has been optimised

217. Each argument is then subdivided into further sub-arguments, which focus in on more detailed aspects of the argument. These largely follow the structure that is set out in the “Holistic Summary”, which I will describe below, and has been used by the RP to provide the narrative to support the arguments and sub-arguments.

218. The Holistic Summary in the Route Map describes the approach the RP has taken to demonstrate the minimisation of radioactivity in plant as follows (Ref. 44):

- Identify key radionuclides of interest, their sources and the main influencing factors.
- Identify the transport mechanisms and the main SSCs and factors influencing those.
- Initially quantify (estimate) and characterise (chemical/physical characteristics) all relevant radioactive species (i.e. determine the source terms).
- Identify the measures that ensure radionuclide generation and transport, and radioactivity levels are prevented/minimised/optimised/managed/controlled from source to disposal.
- Demonstrate all relevant measures have been considered and implemented for the UK HPR1000, with due consideration for proportionality. Propose and implement modifications, if necessary.
- Summarise these steps into a demonstration that risks are reduced SFAIRP.
- Update the source term where required.

219. The RP then presents information about the radionuclides in two sections, according to the two arguments described previously; that the generation of radionuclides has been minimised, and the transport of radionuclides has been optimised. Of note:

- The Route Map describes the various means of minimising the generation of radionuclides, and describes the means to control and minimise the transport of those radionuclides throughout the plant (Ref. 44).
- The Holistic Summary section of the Route Map summarises the key aspects of the activities the RP has undertaken to prevent and minimise the generation of radioactivity, and the transport of any radioactivity that is produced, and generally provides sufficient references to the more detailed information that is available in the wider safety case.
- RO-UKHPR1000-0026 Action 2 required the RP to provide a demonstration that radioactivity has been reduced SFAIRP. The Route Map summarises the actions that have been taken to optimise the primary circuit chemistry regime,

in terms of the reduction and generation of radioactivity. The Route Map adequately explains for each factor why the choice had been made, and why further improvements could not be reasonably made. The RP also highlighted areas where improvements may be possible in the future, should developments in technology materialise, or sufficient OPEX be obtained from other plants. I consider this to be good practice.

220. Although detailed operational parameters and/or controls will be specified by the licensee, it is my expectation that during GDA, the RP is expected to identify the most significant parameters that could affect the generation of radioactivity in the UK HPR1000. The Route Map provides a lot of detail about relevant operating practices related to radioactivity production and transport (Ref. 44). However, there is very little discussion of how these have been optimised. Although the RP identifies in the response to RQ-UKHPR1000-1121 that detailed information about operating practices will be determined in the site-specific stages, I consider this to be an Assessment Finding (Ref. 3).

AF-UKHPR1000-0115 – The licensee shall, as part of developing the operational chemistry safety case, demonstrate how operating practices have been optimised to reduce the risks associated with the generation, transport and accumulation of radioactivity, so far as is reasonably practicable. This should include the activities necessary during start-up and shutdown of the plant.

221. The Route Map provides an adequate description of the primary circuit chemistry during normal operation, and also describes how this has been optimised to reduce the generation and accumulation of radioactivity (Ref. 44). However, the description of the chemistry choices during start-up and shutdown does not describe how these actions have been optimised to reduce radioactivity SFAIRP. I consider this to be part of Assessment Finding AF-UKHPR1000-0115.

222. The Route Map describes the materials choices and their surface treatments, and accurately reflects the position reached in other parts of the generic safety case (Ref. 44). However, the proposed selection process for the hard facing materials in valves (retention of the high cobalt-based alloy, Stellite™ or replacement with an alternative, low cobalt alloy) does not consistently follow the hierarchy of risk control (Eliminate, Reduce, Isolate, Control) for all valves, nor does it consistently follow RGP. Categorisation of valves according to risk of cobalt release has been presented to support this deviation away from RGP, without sufficient evidence of how these risks were determined and therefore I consider that the RP has not adequately demonstrated that risks have been reduced SFAIRP. Materials choices are discussed in sub-section 4.2.3.1 of this report, and this has been captured as AF-UKHPR1000-0113.

223. The third action of RO-UKHPR1000-0026 required the RP to demonstrate that the generic UK HPR1000 design is capable of minimising radioactivity in the primary circuit and connected systems. This action required the RP to demonstrate that the clean-up systems in the UK HPR1000 have adequate capability and capacity to manage the levels of radioactivity in the plant. Part of the work required to close this action is also assessed in sub-section 4.2.2 of this report, which focusses on the supporting systems to the primary circuit, and includes several systems that are used to clean-up the primary circuit. See also the assessment of the TEG [GWTS] and the Liquid Waste Treatment System (TEU [LWTS]) in sub-section 4.4.3.

224. As discussed in sub-section 4.2.2 of this report, during Step 4 of GDA I employed a TSC to review certain aspects of the generic safety case, which included the adequacy of the systems connected to the primary circuit, such as the RCV [CVCS] and the TEP [CSTS]. These systems have an important role in the reduction of radioactivity in the

primary circuit, especially the RCV [CVCS]. My TSC concluded that the justification of the relevant systems was generally appropriate, and they had confidence that they would function as required. However, the RP did not provide OPEX in the form of operational plant data to supplement the theoretical analysis that formed the basis of much of the system substantiation. This forms part of Assessment Finding AF-UKHPR1000-0011.

225. The fourth action of RO-UKHPR1000-0026 required the RP to identify any controls necessary to ensure radioactivity in UK HPR1000 will be minimised, which would include any limits and conditions necessary in the interests of safety. The Route Map clearly provided links to relevant limits and conditions necessary in the interests of safety from within the Chemistry topic (Ref. 44). Many of the limits and conditions necessary in the interests of safety derived within the generic safety case have been aimed at the reduction of corrosion, either via operating practices, or the prevention of the occurrence of impurities within the circuit, and I am generally content that they are reasonable.

226. The final action of RO-UKHPR1000-0026 concerns the analysis of CPs, and is discussed in sub-section 4.2.3.3. I will assess the RP’s approach to the minimisation of fuel clad corrosion in sub-section 4.2.5. However, despite the Assessment Findings identified above, I judged that the RP had provided sufficient evidence to close the RO.

227. In terms of the control of primary circuit water chemistry, the RP outlines the main measures that have been adopted to maintain safe operation and minimise the generation and transport of radioactivity (Ref. 44). These are:

- Optimisation of the lithium concentration for optimal pH control – a target $pH_{300\,°C}$ of 7.2.
- During start-up, oxygen is removed by hydrazine dosing in order to minimise the risk of an oxidising environment that would present a challenge to the integrity of materials. Oxygen concentrations are controlled during operation by the addition of hydrogen.
- Impurity control to reduce corrosion and scaling of fuel cladding.
- Application of zinc injection which reduces general corrosion rates, reduces incorporation of ^{60}Co into the SSCs out of the core, and consequently reduces CPs transport to the fuel.

228. The approach taken in terms of pH control is assessed in the following sections. The addition of hydrogen to minimise concentrations of oxygen during normal operations is assessed in sub-section 4.2.7. The addition of zinc, in an effort to reduce corrosion rates and radioactivity is discussed in sub-section 4.2.6.

Corrosion Products

229. In sub-section 4.2.3.1 I assessed the RP’s approach to material selection to minimise the generation of CPs SFAIRP. Although the RP has outlined how CPs will be minimised by the generic design, operation and chemistry of the UK HPR1000, the only estimation of the generation and transport throughout the plant is made in the derived source term document, which is largely based upon OPEX from the fleet of existing CGN operated plants (Ref. 67). Although these plants are similar to the UK HPR1000, there are several plant-specific aspects of the design that could affect the production and transport of CPs. The RP had provided an analysis of the amount of CPs in the primary coolant to address one aspect of RO-UKHPR1000-0015, and so I asked the RP how this information would be used to provide a quantitative estimate of the CPs in the UK HPR1000 in RQ-UKHPR1000-0995 (Ref. 4, Ref. 3). Following an inadequate response, I amended RO-UKHPR1000-0026 to include an additional action (Ref. 4). This action required the RP to provide an estimate of the generation and

transport of CPs, based upon the design and operation of the UK HPR1000, and to justify that they had been reduced SFAIRP.

230. In response to the new action under RO-UKHPR1000-0026, the RP identified that the fuel deposit modelling code CAMPSIS would be used to produce a quantitative estimate of the generation of CPs (Ref. 4). My assessment of the adequacy of CAMPSIS is detailed in sub-section 4.2.5, but I am content that it is an adequate tool to provide an estimate of the generation of CPs in the UK HPR1000.

231. The RP produced a new deliverable ‘The Corrosion Product Source Term Analysis with UK HPR1000 Specific Design’, which describes the chosen method, and presents the results of the work to quantify the generation and transport of CPs (Ref. 45). To estimate the generation of CPs, the RP grouped the radionuclides into two separate groups, and applied different methods for each. This is largely due to the availability of relevant corrosion data for the two groups. For both of the following groups, CAMPSIS is used to estimate the amount of these species entering the coolant, using plant-specific factors such as wetted surface area and the primary circuit chemistry. The two groups are as follows:

- The first group comprises ^{59}Fe and ^{51}Cr, and CAMPSIS assumes that a proportion of the species within the structural materials are lost to the coolant, based upon OPEX from other plants.
- The second group comprises ^{60}Co, ^{58}Co and ^{54}Mn. The RP conservatively assumes that all of these species present within the metals are lost to the coolant.

232. As I described in the RO-UKHPR1000-0026 closure note, CAMPSIS firstly calculates the coolant activity using plant-specific parameters, and is then scaled against the appropriate values from CGN CPR1000 plants, in order to provide a best estimate analysis that partially reflects the impact on the source term of the differences in design and chemistry between the UK HPR1000 and the CPR1000 (Ref. 74). I am content with this approach for GDA, but this topic should be developed by the licensee to gain a deeper understanding of the role that CPs will play once the reactor is operational. As CPs are a significant component of the minimisation of radioactivity, I consider this to be an Assessment Finding.

AF-UKHPR1000-0117 – The licensee shall develop the plant-specific corrosion product estimations, including all relevant factors, to justify that risks have been reduced so far as is reasonably practicable. The analysis should assess how the coolant chemistry and the plant design and operation, affect the generation and transport of all relevant corrosion products.

233. A related assessment finding has also been raised by my Radiological Protection colleagues, AF-UKHPR1000-0096 (Ref. 36).

234. Whilst the assumption described above (that all cobalt from stainless steels and 690TT is released into the coolant) is conservative for those materials, a significant contributor in the cobalt source term is from materials which have a high cobalt content, for example Stellite^{TM}, which is assessed in more detail in sub-section 4.2.3.1. I requested clarification about how the role of cobalt had been considered within the analysis (RQ-UKHPR1000-1668 (Ref. 3)). The RP stated that as CAMPSIS was designed to estimate fuel deposits, and that cobalt is a minor contributor to fuel deposits, an OPEX-based approach for cobalt in the CP source term analysis had been adopted instead. This may be a reasonable approach, but the RP did not clarify how high cobalt-bearing materials had been considered. The approach to cobalt in this analysis had not been adequately justified. This is part of Assessment Finding AF-UKHPR1000-0112.

235. The results for the first group of radionuclides show a small reduction in the production of CPs when compared to CGN PWRs, but similar generation profiles are produced. For activated CP generation in the coolant (or coolant activity), a rapid increase is predicted during first operation, followed by a gradual decline. For both coolant and deposited activity, a rapid increase during first operation is observed; the rate of which gradually decreases.

236. The RP goes on to describe a third group of radionuclides, which comprises ^{110m}Ag, ^{122}Sb and ^{124}Sb. The RP describes the behaviour of these species as complex and notes that it cannot be reliably calculated by CAMPSIS, and therefore an OPEX-based approach is adopted. Any design improvements included in the UK HPR1000 (compared to CPR1000), in terms of a reduction in the generation of these nuclides (generally from reductions in wetted surface areas of components), is factored into the analysis. In my assessment of the closure of RO-UKHPR1000-0026, I judged that this method was a reasonable one for GDA (Ref. 74). However, the evidence to support the average values used for these species is not reported in the safety case. This is related to Assessment Finding AF-UKHPR1000-0011.

237. RO-UKHPR1000-0026 also asked for the RP to provide analysis of the transport of CPs to other systems associated with the primary circuit, such as the IRWST and the RCV [CVCS] (Ref. 4). The deliverable ‘The Corrosion Product Source Term Analysis with UK HPR1000 Specific Design’, provides a discussion of the results which will aid users of the case, and a justification that the risks from CPs in those systems have been reduced SFAIRP (Ref. 45). I am satisfied that the work set out in the analysis is sufficient.

238. All of the matters raised as part of the closure of RO-UKHPR1000-0026 have been dealt with in this report; I raised six matters, which have been captured as part of five Assessment Findings, (AF-UKHPR1000-0011, -0112, -0115, -0116 and -0117).

Effects of Chemistry Controls on Radioactivity

239. The RP has described how the primary circuit chemistry of the UK HPR1000 has been designed in order to minimise the generation of CPs:

- pH control strategy – a target pH value of 7.2 is set to reduce the impact on radioactivity caused by general corrosion. EBA is used to allow this pH control strategy;
- reducing environment control – hydrogen is added to the primary circuit, to suppress the oxygen species generated by flux radiolysis and to maintain the reducing environment to minimise corrosion of structural materials – the addition of hydrogen to the primary circuit is assessed in sub-section 4.2.7;
- passivation during HFT – the proposed HFT programme is designed to minimise CP release by passivation (formation of a protective surface film) of components and pipes during commissioning, described in sub-section 4.2.11; and
- zinc injection will be applied in the UK HPR1000, and the use of zinc will reduce the corrosion rate. The proposals for zinc injection are assessed in sub-section 4.2.6.

240. The RP has chosen a target normal operating pH of 7.2, and a normal operating range between 6.9 and 7.4. The RP has justified this target pH and operating range using information from the following sources:

- a review of the common corrosion mechanisms affecting the materials used within the primary circuit and connected systems;
- literature reviews of international studies, guidelines and good practice; and
- OPEX from Chinese plants, as well as other new build PWR plants.

241. I am content that a target pH of around 7.2 represents current good practice, following a review of international guidelines, which includes advice from EPRI, EDF and Verenigate Grosskraftwerke Betreiber (VGB) (Ref. 75).

242. To enable this chemistry regime, the RP will use lithium hydroxide (enriched in ^{7}Li at 99.9%), and EBA (enriched in ^{10}B). The concentration of boric acid will vary throughout the cycle, starting the cycle with a high concentration and gradually decreasing as the number of fissile nuclei in the fuel decreases (fuel burn-up). There is a limit to the lithium concentration (of 3.5 mg kg^{-1}), that can be used, which is due to the risk of corrosion of the fuel clad, and the fuel supplier has specified such a limit for UK HPR1000 (Ref. 76).

243. Theoretically, a higher pH could be achieved using a higher enrichment of boric acid, whilst respecting the lithium concentration limit required by the fuel vendors. However, as well as a significant increase in cost, this change would also have impacts on fuel management. The RP has argued that such an increase would represent a significant change to the design of the plant, and would not be practicable. Given the adherence I have noted with international guidance, I am content that an increase in enrichment would not give a sufficient benefit to be practicable.

244. As well as minimising the amount of corrosion, and therefore the quantity of CPs entering the primary circuit, the pH regime should also maximise the solubility of those CPs, for them to be efficiently removed by the RCV [CVCS] during normal operation. Following a request for more information about this topic in RQ-UKHPR1000-1300, the RP provided a review of the solubility of the CPs, and information about particulate sizes where the CPs are not soluble in the primary coolant, based on OPEX from a number of sources (Ref. 3). The response reinforced the claim that the pH regime is one which seeks to reduce CP generation and dissolution during normal operations as much as possible, and to remove as many CPs as possible during shutdown. The response also provided details of the expected particle size distribution, based upon OPEX from plants using 690TT SGs. I was satisfied that the RP had adequately considered these aspects in the choice of pH regime.

245. With such a limit on lithium concentration, the use of natural boron would not allow a pH of 7.2 to be reached at the start of the cycle; a much higher boric acid concentration would be required due to the lower proportion of ^{10}B in natural boric acid. With the use of EBA, UK HPR1000 can operate with a constant pH “coordinated chemistry” regime for most of the cycle; the EBA enrichment determines the pH at the BOC. In this type of regime, the pH at temperature ($pH_{300\ °C}$) is maintained constant and the lithium concentration is balanced with the reducing boron content throughout the cycle. During Steps 3 and 4 of GDA, the RP’s documentation offered differing views on the pH at the BOC, when the boron concentration is at its greatest. In RQ-UKHPR1000-0709, I asked what the expected pH would be at the BOC, and if it was not 7.2, what the consequences would be for the materials of the primary circuit (Ref. 3).

246. In response, the RP provided calculations (using the chemical expressions and equilibrium constants provided in EPRI Primary Water Chemistry Guidelines), which show the pH obtained at the start of the first few cycles, and during an equilibrium (Ref. 76). In summary:

- For Cycle 1 and Cycle 2 with lower total boron concentrations, pH 7.2 is established at BOC and a constant pH of 7.2 is maintained throughout the cycle.
- For Cycle 3, due to a relatively higher boron concentration at BOC, the pH value does not reach 7.2 at first, but is gradually increased with the increase of burnup and decrease of boron concentration, (an initial pH of 7.14 increases to 7.18 within 60 hours, and then to 7.2 within a further 24 hours, based upon CPR1000 OPEX).

247. Although the pH does not reach 7.2 for a short time following start-up, I am content that the time with a lower pH is relatively short, and would not present a significantly increased risk of corrosion to the plant.

Activation Products and Other Radionuclides

248. Some radionuclides are produced through the activation of species in the primary coolant by the neutron radiation field. Some of these activation products are unavoidable, such as nitrogen-16 (^{16}N) and ^{14}C, but many can be controlled and minimised.

249. During reactor operations, ^{16}N is the dominant nuclide in the coolant. Due to its high activity and highly energetic gamma decay (around 6 MeV), access to the RCP [RCS] is restricted and shielding around the RCP [RCS] is required. However, with a half-life of only 7 seconds, the activity follows the reactor power (specifically neutron radiation) almost instantaneously, so it is not an issue at shutdown. ^{16}N is used as an indicator of SG tube leaks via gamma detectors in the main condenser off gas system. Argon-41 (^{41}Ar) is of little radiological consequence but is used as an indicator for air ingress to the primary circuit.

Tritium

250. ^{3}H is a radioactive by-product of water moderated and cooled nuclear reactors which can contribute to ORE and is an important nuclide subject to environmental control. ^{3}H is, and has been, a key feature in determining the shutdown profile in a number of PWRs. It is mainly produced by neutron bombardment of boron and lithium and also escapes slowly from fuel pins, SNSs and some control rods.

251. The RP produced a report that discusses the main sources of ^{3}H that enter the primary circuit, and any possible routes by which it can be reduced (Ref. 77). There are five areas that the RP has identified that affect the production of ^{3}H:

- ^{3}H is generated by fission of the nuclear fuel and a small percentage diffuses through the cladding of the fuel into the coolant. The reference design uses M5 Zircaloy cladding, and I am content that this material will reduce transport of ^{3}H SFAIRP.
- ^{3}H is produced via alpha activation of ^{10}B in the coolant. By using EBA, less lithium hydroxide is required to be added to the coolant to reach the target pH, which has a beneficial impact on the amount of ^{3}H that is generated.
- The RP claims that the use of burnable poisons (gadolinium oxide in the fuel pellets), means that less boron is required to maintain reactivity control and pH control, which achieves a 30% reduction in the required boron concentration, and a similar reduction in ^{3}H production from this mechanism. The amount of burnable poisons that can be used is limited, since they degrade the thermal conductivity of the fuel pellets.
- SNS are included within the core to provide a measurable background neutron count-rate for the core detectors. The generic UK HPR1000 design proposes the use of antimony-beryllium sources, which is the material used in many other plants worldwide. Beryllium-9 (^{9}Be) in the SNS generates significant quantities of ^{3}H via the two step reaction $^{9}Be\ (n,\alpha)\ ^{6}Li\ (n,\alpha)\ ^{3}H$. The RP estimates that as much as 46% of ^{3}H is via the SNS (Ref. 78). The RP states that a Zircaloy-type clad of the SNS could potentially reduce the escape of ^{3}H. However, this type of clad material has never been used in PWRs, and the lifetime of Zircaloy is much shorter than stainless steel, owing to irradiation effects and corrosion behaviour. Considering cost, generation of waste and disposability, the RP states that stainless steel cladding is the choice that reduces risks SFAIRP. The RP suggests that OPEX from the reference plant (FCG3) may allow a future assessment to be made and possible reductions in

the mass of antimony-beryllium used in the SNS. I note that the feasibility of changes to the SNS proposals for the UK HPR1000 is being evaluated and managed in PCER Chapter 3 as a forward action (Ref. 78). I consider this to be a reasonable proposal, and is for the licensee to consider in due course.

252. I am content that the RP has adequately demonstrated that the generation of ^3H has been reduced SFAIRP.

253. Although the source term documents provide an estimate for ^3H generation in the primary circuit, this is based upon the use of OPEX from Chinese CPR1000 plants, which are of a different design. Nor do these documents provide an estimation of the transport and distribution of ^3H from the primary circuit to other areas of the plant. I therefore raised RO-UKHPR1000-0049 (Ref. 4).

254. The RP's response to RO-UKHPR1000-0049 involved the production of a new document, 'Identification and Minimisation Demonstration of Tritium Related Safety Risk', which introduced four main topics (Ref. 79):

- a route map of the whole life cycle of ^3H in the plant;
- the identification and implementation of key controls;
- a demonstration that ^3H activity levels have been reduced SFAIRP; and
- an analysis of the ^3H radiological risk on site.

255. I will discuss each of these in the proceeding sections. In addition to this new deliverable, the RP also updated several documents that are part of the source terms safety case, to include details of the method and the calculation of ^3H concentrations in the primary circuit and associated systems.

256. The Route Map provides a more detailed description of the main sources of ^3H than had been available previously (Ref. 79). It also describes the various means with which it is transported from the primary circuit to the other systems associated with the primary circuit, including a consideration of the two main methods that the RP has identified to manage ^3H within an operating cycle; continuous dilution and end of cycle dilution. Finally, the Route Map considers how ^3H will be discharged from the plant, and provides references to where this aspect is considered in more detail (Ref. 79).

257. The general approach to ^3H is described as a Tritium Management Strategy (TMS), and the RP has provided details of several methods that a future operator could follow in the control of ^3H (Ref. 79). Two main strategies have been proposed:

- An end-of-cycle dilution strategy, where ^3H continuously builds up in the primary circuit during a cycle, and is discharged before shutdown to ensure a head-lift criterion is met (prior to the top of the reactor vessel being removed, a limiting ^3H concentration is often applied).
- A continuous dilution strategy, based upon a target value of ^3H activity in the primary circuit. This is also known as bleed and feed.

258. The RP identifies that when defining the TMS and associated limits, a balance between the reduction of the ^3H-induced radiological dose and environmental impact, worker burden, the worker doses from ^3H dilution operations and measurement, and economic benefits should be achieved. The RP also describes some approaches taken by other operators, including at Sizewell B in the UK. The RP stated that the choice of TMS, and any limits associated with ^3H are for a future operator to determine.

259. The RP has identified the need for a ^3H limit at the end of a cycle prior to the removal of the vessel head, "to agree with international good practice" (Ref. 79). However, the RP has not identified a numerical limit during GDA. To explain this position, the RP has conducted a series of calculations to understand the role ^3H concentrations have on

ORE. Using a value that is greater than the maximum expected concentration of ^{3}H in the primary circuit, the RP has estimated that the largest effective operator dose rate is in the range of several µSv hr^{-1} or below, and the RP states that as such, it is very low (Ref. 66). This analysis accounts for multiple fuel cycles, and includes systems such as the SFP, and IRWST.

260. I asked a Radiological Protection colleague to review this aspect of the resolution work, and they were satisfied with the method that the RP followed and the RP's interpretation of the results (Ref. 80).

261. Although the RP has evaluated the contribution of ^{3}H to ORE, and has indicated a likely range for a future ^{3}H limit, my expectations were that a limit would be identified during GDA. I therefore consider this to be an Assessment Finding.

AF-UKHPR1000-0118 – The licensee shall, as part of developing the operational safety case, define limits and conditions to ensure that the risks associated with tritium have been reduced so far as is reasonably practicable.

262. To complement the qualitative description provided in the Route Map, the RP also provided an analysis of ^{3}H generation rates within the primary circuit of the UK HPR1000 (Ref. 79). In my assessment of the closure of RO-UKHPR1000-0049, I judged that the calculation of ^{3}H in the primary circuit was adequate, as described in (Ref. 66, Ref. 81, Ref. 67, Ref. 82). However, in the method described by the RP, the quantity of ^{3}H derived from SNS was based upon Chinese OPEX that was not shared as part of the case. As SNS account for about half of the calculated level of ^{3}H, I consider that the basis of this data should form part of the licensee's safety case. As such, I consider this to be part of Assessment Finding AF-UKHPR1000-0011 on OPEX**.**

263. The second aspect of RO-UKHPR1000-0049 that I assessed was the description of operator actions that may affect the transport of ^{3}H in the UK HPR1000, and the identification of relevant controls. As part of this work, the RP has provided a coherent, detailed description of the activities where operator actions may impact the transport and accumulation of ^{3}H, and ranked these factors on the basis of how significant they are. I judged the aspects associated with the evaporation of ^{3}H from open bodies to be reasonable. However, the discussion of the role that coolant recycling has in the accumulation of ^{3}H was based upon qualitative arguments that were not supported by plant data. Although the arguments appeared to be reasonable, I consider that quantitative data would support the demonstration that the risks from ^{3}H have been reduced SFAIRP. This is part of Assessment Finding AF-UKHPR1000-0118.

264. Despite these findings, my assessment of the closure of RO-UKHPR1000-0049 means that I am content that the RP has now produced an adequate safety case to demonstrate that the risks associated with ^{3}H are understood, and have been reduced SFAIRP. All of the matters raised as part of the closure of RO-UKHPR1000-0049 have been dealt with in this report; I raised three matters, which have been captured as part of two Assessment Findings, (AF-UKHPR1000-0011 and -0118).

Carbon-14

265. There are two main sources of ^{14}C in the UK HPR1000:

- from oxygen-17 (^{17}O) which is naturally present in the coolant; and
- from nitrogen dissolved in the coolant, from the use of nitrogen as a cover and flushing gas in the VCT, and the REA [RBWMS].

266. The UK HPR1000 source term, states that UK HPR1000 will produce 330 GBq of ^{14}C per year (Ref. 66). Of this, 88% comes from the ^{17}O reaction, with the remaining 12%

from ^{14}N dissolved in the coolant. To calculate this figure, the RP has used an assumption of 18 ppm nitrogen in the primary circuit (Ref. 81). I note that this is the same method and assumptions that have been used by other RPs in GDA, and I am satisfied that it is reasonable.

267. Other reactors in the past have used hydrogen as a cover gas in gaseous waste processing systems, which has the advantage that it does not act as a source of ^{14}C. The RP has estimated that the use of nitrogen will result in an increased production of ^{14}C by about 11%. However, there are other hazards associated with the use of hydrogen, such as the risk of deflagration. The RP has acknowledged the increase in ^{14}C, and presented arguments to demonstrate how arisings of this species have been minimised (Ref. 44). On balance, despite the increased ^{14}C arisings, I am content that the use of nitrogen is the option which reduces risks SFAIRP, due to the significant safety improvements from the removal of significant quantities of hydrogen from the design. This topic has also been reviewed by the Environment Agency and Nuclear Liabilities Regulation colleagues and judged to be adequate (Ref. 69, Ref. 68). The impact that dosing DZA into the primary circuit has on ^{14}C production is considered in 4.2.6.2.

4.2.3.4 Radiochemistry and Radiochemical Parameters

268. In addition to the inactive chemical species that are described in the safety case, the designer of a PWR should describe the control of various radiochemical parameters in the primary circuit, and other auxiliary circuits within the safety case. There are numerous reasons why certain radiochemical species should be monitored during a fuel cycle:

- monitoring of radioactive species prior to shutdown, for worker dose purposes;
- monitoring of fission products and other species to provide an adequate failed fuel monitoring system;
- monitoring of ^{16}N in order to detect any primary to secondary SG leakage; and
- compliance with initial conditions of accident scenarios of radiological consequence studies, for example the sum of noble gases, and equivalent iodine-131 (^{131}I).

269. At the beginning of Step 4 of GDA, none of the main chemistry safety case documents contained any information about the control or monitoring of the various radiochemistry parameters. During Step 4, a new deliverable was produced, ‘Radiochemistry Parameters Value’ (Ref. 31). This document was more than just a list of radiochemical species to be controlled or measured; it also described why certain species should be monitored to ensure that failed fuel can be detected.

Integrity of Fuel Cladding and Failed Fuel Detection

270. In PWRs, the sources of radioactivity include fission products, CPs, activation products and actinides, among which fission products and actinides are normally retained inside the fuel cladding. However, they can be released into the primary coolant as a result of fuel failure. Tramp uranium on fuel cladding surfaces can also be released into the primary coolant during plant operation. Within the cavity of an intact fuel rod, various species are distributed:

- Noble gases
- Xenon-133 (^{133}Xe), the most abundant fission product isotope
- ^{131}I, which is the most abundant of the iodine isotopes

271. When a fuel failure occurs, the escape of fission products and actinides is controlled by their chemical and physical characteristics. Fission gas release is controlled by

diffusion and clad defect size. The release of other fission products and actinides is influenced by other factors, such as solubility, volatility and chemical affinity.

272. The RP states that the objective of fuel failure detection is two-fold:

- Monitor and detect the potential failure of fuel cladding, and respond in a timely manner to avoid further deterioration of the fuel.
- Limit radioactive substances from the release, and minimise pollution and radiation exposure to personnel, systems, equipment and the environment.

273. For the purposes of this assessment, I sought to gain confidence that appropriate species had been selected for failed fuel detection, and that appropriate limits had been selected to successfully meet the objectives described above.

274. For the onset of a fuel cladding defect, the RP identifies two separate categories of indicators; noble gases and iodine. The RP has chosen the sum of noble gases and the amount of iodine equivalent to ^{131}I, as these parameters provide input data to the system to warn of failed fuel and are used for radiological consequences analysis. ^{131}I has been chosen due to its long half-life and radiotoxicity.

275. The parameters chosen by the RP (the sum of noble gases and equivalent iodine) are commonly provided by PWR operators as the main means of detecting the onset of a fuel failure, and therefore I consider them to be a reasonable choice. However, the RP initially provided very limited information in terms of a limit, and a justification of why that limit is reasonable. In response to RQ-UKHPR1000-1449, which sought further clarification on why these limits were appropriate for their purpose, the RP stated that due to difficulties in determining the volume activity of iodine and noble gases via a theoretical means (due to variability of fuel failure size and the location of failure), a method has been used to determine the limits that is based upon CGN operating data (Ref. 3). The RP states that the equivalent iodine limit is slightly higher than the maximum measured iodine activity in CGN units. I raised a further RQ (RQ-UKHPR1000-1693) in order to gain a better understanding of the basis of this important limit (Ref. 3).

276. In response to this RQ, the RP set out a series of reasons why a lower primary circuit activity limit would not reduce risks SFAIRP overall. This included factors such as increased operator burden, waste arisings, spent fuel arisings, water usage and decreased economic efficiency. The RP also described the approach that is taken at a number of other facilities that are either operating, or planned to operate. This shows that an equivalent iodine limit for the UK HPR1000 of 16 GBq t^{-1} is similar to the other plants reviewed. I am therefore satisfied that the RP has demonstrated that this limit is appropriate for the current stage of generic safety case development. I have also assessed the implications for this limit during faults; this can be found in sub-section 4.5.3.1.

277. The Radiochemical Parameters Value document also provides a limit for iodine-134 (^{134}I) to provide information on the amount of fuel dissemination, and references an IAEA document, which states that a limit of [redacted] GBq t^{-1} is used to limit the dissemination of new fuel material in the primary coolant (Ref. 31, Ref. 83). However, the RP then states that a limit of [redacted] GBq t^{-1} should be adopted for the UK HPR1000. The response to RQ-UKHPR1000-1449 did not provide any more information about why a [redacted] GBq t^{-1} limit is adequate (Ref. 3). I consider this to be a minor shortfall.

278. The RP suggests that a limit of [redacted] for the ^{133}Xe/^{135}Xe ratio, and a limit of [redacted] MBq t^{-1} for ^{133}Xe would be adequate to provide appropriate information on the onset of a fuel failure; these limits are based upon suggested limits provided by IAEA, and I am content that they are reasonable. The proposed radiochemical parameters are shown in Table 6.

Table 6: Radiochemical Parameters

Species	Operating Window (MBq t^{-1})	Limit (MBq t^{-1})	Operating Mode
^{58}Co	[redacted]	[redacted]	Reactor pool flooding completed
	[redacted]	[redacted]	Before RPV opening
	[redacted]	[redacted]	From oxygenation to last main shutdown pump
^{133}Xe	[redacted]	[redacted]	Before RPV opening
	[redacted]	[redacted]	From RIS [SIS] / RHRS connection to oxygenation
	[redacted]	[redacted]	From oxygenation to last main shutdown pump
^{131}I	[redacted]	[redacted]	Before RPV opening
	[redacted]	[redacted]	From RIS [SIS] / RHRS connection to oxygenation
	[redacted]	[redacted]	From oxygenation to last main shutdown pump
Total gamma	[redacted]	[redacted]	Reactor pool flooding completed
	[redacted]	[redacted]	Before RPV opening
	[redacted]	[redacted]	From oxygenation to last main shutdown pump

4.2.3.5 Strengths

279. The generic safety case has identified topics that may be developed in future safety cases (such as ^{3}H limits and the use of SNSs), and has suggested that these be reviewed in the site-specific stages. I consider this to be a strength of the case.

280. During Step 4 of GDA, the RP successfully developed the generic safety case to produce an adequate demonstration that radioactivity has been reduced SFAIRP. This is a broad topic, with many interconnected factors, but the RP has clearly identified why risks could not be reduced further, and therefore why risks associated with the generation and transport of radioactivity are reduced SFAIRP.

4.2.3.6 Outcomes

281. Based upon the assessment of the minimisation of radioactivity in UK HPR1000 described in sub-section 4.2.3 above, I have identified several Assessment Findings which need to be addressed by the licensee. These concern justifying that the cobalt inventory within UK HPR1000 has been optimised (AF-UKHPR1000-0112), a justification of which valves will include cobalt-based hard facings (AF-UKHPR1000-0113), a demonstration that the manufacturing routes of key SSCs reduce radioactivity

SFAIRP (AF-UKHPR1000-0114), a demonstration of how operating practices have been optimised to reduce the risks associated with the generation, transport and accumulation of radioactivity SFAIRP (AF-UKHPR1000-0115), a development of the plant-specific corrosion product estimations, to justify that risks have been reduced SFAIRP (AF-UKHPR1000-0117) and a definition of the limits and conditions to ensure that the risks associated with ^3H have been reduced SFAIRP (AF-UKHPR1000-0118). I have also identified a number of areas that are relevant to the Assessment Finding concerning inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011), raised earlier in this report.

282. I also identified three minor shortfalls as discussed in sub-section 4.2.3 above.

4.2.3.7 Conclusions

283. I am content that the RP has produced an adequate safety case to justify that the generation, transport and deposition of radioactivity has been minimised, including an adequate justification that the choice of materials, the primary circuit chemistry regime and supporting systems support the claims made in the case. Although I have raised a number of Assessment Findings, I am content that the RP has developed the safety case in this area sufficiently for this stage of the development of the generic design of the UK HPR1000.

284. Based on the outcome of my assessment of the minimisation of radioactivity, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

285. I have identified six Assessment Findings as a result of this part of my assessment.

286. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the generic safety case.

4.2.4 Material Integrity and Corrosion Control

287. A number of key SSCs of the primary circuit cannot, or will not, be replaced over the lifetime of the reactor (a key example is the RPV). These SSCs must maintain their integrity over the reactor lifetime; in other words, they must retain sufficient strength to retain pressure. From a Chemistry perspective, corrosion to a sufficient degree over the reactor life may decrease this strength to a point where damage, such as leaks or failures, occurs.

288. Corrosion may begin from both the inside of the reactor (as a result of poor primary coolant chemistry control) and the outside of the reactor. Areas that might be of concern for corrosion include:

- Heat exchanger tubing that acts as a barrier between the radioactive primary circuit and the nominally clean circuits (secondary and auxiliary circuits), particularly the SG tubing for which it is harder to isolate a leak.
- The main pipework of the coolant circuits that retain pressure.
- Internal pipework and components, such as the control rods and safety valves, for which alignment and functional integrity are important.
- The cladding that protects the Low Alloy Steel (LAS) of the RPV and other components from the coolant.

289. General corrosion from the primary coolant is slow and easily detectable in modern PWRs due to the alloys selected, and so is not a threat. However, localised corrosion phenomena, such as SCC or Intergranular Attack (IGA) may occur that would be a concern if not detected in time. The risk of these types of corrosion can be reduced through selection of compatible materials and coolant chemistry:

- SCC requires a combination of tensile stress, corrosive environment and a susceptible material to occur. In PWR primary coolant, this is referred to as Primary Water SCC (PWSCC) or Irradiation-Assisted SCC (IASCC) when the effects of irradiation on the material are a factor.
- IGA requires a combination of a corrosive environment and a susceptible material to occur. It occurs along or near grain boundaries due to segregation of impurities or enrichment/depletion of an alloying element in the grain boundary region which arises from a sensitising heat treatment.

290. General corrosion can occur on external surfaces in the presence of concentrated boric acid. This environment can occur when boric acid is transferred from the primary coolant by a leak and then concentrated by evaporation (such as has occurred at Davis Besse in the USA (Ref. 84)). It is mainly an inspection matter and was covered briefly in my assessment. In this section I mainly assess the potential for cracking of the reactor from the inside, caused by primary coolant. Chemistry affecting the integrity of the fuel cladding is described in sub-section 4.2.5.3.

4.2.4.1 Material Selection for Corrosion Prevention

291. The RP has selected the SSCs that are identified as vulnerable to SCC, IASCC, or Flow Assisted Corrosion (FAC) for detailed assessment as part of GDA. I consider this an appropriate approach, noting that certain component material choices will require further justification in the site-specific stages as part of normal business.

292. The RP makes the following sub-claim on material selection for corrosion prevention (Ref. 5):

- 3.3.10.SC21.1: The primary chemistry and process are optimised in all operating modes to maintain the integrity of the safety barriers in the primary circuit.

293. The main reports detailed in the PCSR are the ‘Material Selection Summary Report’, ‘Ageing and Degradation Justification Summary Report’, ‘Topic report on Surface Treatment of SSCs’ and ‘Operating Experience with AFA 3G™AA Fuel Assemblies’ (Ref. 48, Ref. 57, Ref. 58, Ref. 85). The material selection and corresponding ageing and degradation reports of the high integrity components have also been detailed separately for the MCL (Ref. 86, Ref. 87), PZR (Ref. 88, Ref. 89), RCP (Ref. 90, Ref. 91), RPV (Ref. 92, Ref. 93), RVI (Ref. 94, Ref. 95), and SG (Ref. 56, Ref. 59).

294. Materials selection requires a balance between structural integrity, cost and minimising radioactivity. The structural integrity aspects have been assessed elsewhere (Ref. 96). The radiochemical aspects of material selection are discussed in sub-section 4.2.3.

295. In Step 3 of GDA I examined the RP’s approach to material selection methodology and sampled the material selection for the neutron absorber material. In Step 4, I reviewed the main materials choices presented in the PCSR (Ref. 5). My assessment also sampled the specific material selection and ageing and degradation submissions related to the materials of the SGs in contact with the primary coolant. For this I also reviewed the ‘Material Specification for Steam Generator Tubing Nickel-Chromium-Iron’ (Alloy 690) (Ref. 56). The SGs make up over 80% of the pressure boundary area of the primary circuit and is an area where OPEX of older designs has noted a number of weaknesses that result in degradation. I raised two RQs, RQ-UKHPR1000-1633 and

RQ-UKHPR1000-1640 to gain further clarification and evidence to support the claim made in the PCSR (Ref. 3).

296. My assessment of the SG materials in contact with the secondary coolant and its associated ageing and degradation is covered in sub-section 4.3.4.1 of this report.

297. The principal SAPs relevant to my assessment of the materials integrity are ECH.1, EAD.1 to EAD.2, EMC.13 and EMC.16.

298. Within each of the ageing and degradation reports, the RP placed material selection and manufacturing controls as mitigations to reduce the risk to integrity SFAIRP, whilst control of chemistry and ISI were claimed in managing any remaining risk. This met with my expectation that significant demands should not be placed on chemistry control; rather, chemistry should supplement the materials choices, in keeping with the principle of the hierarchy of control measures to manage the risk SFAIRP.

299. The major materials of construction in contact with the primary circuit of UK HPR1000 are detailed within sub-section 4.2.6 of the PCSR (Ref. 5). The materials selected are primarily stainless steels and nickel-based alloys. The vessels (PZR, RPV and SG) and some components are fabricated from a LAS. Surfaces which are exposed to the primary coolant are clad with nickel-based alloy or stainless steel. These material groups, whilst having good resistance to general corrosion and FAC, may be susceptible to SCC, IGA, pitting and crevice corrosion.

Material Choices

300. This subsection assesses the RP's material choices, for susceptibility to IGA and SCC.

301. The majority of the nickel-based alloys selected are Alloy 690TT. This alloy is selected for the SG tubes, the CRDM adaptor sleeve, and the clevis inserts and positioning pin, and radial support key insert pin plug of the RVI.

302. As part of my assessment, I sampled the RP's assessment of potential material choices of the SG tubes. The preliminary options for the SG tubes were Alloy 690TT and Alloy 800; no other candidate materials were taken forward for consideration (such as Alloy 600 mill annealed, Alloy 600TT, 18-8), all being discounted as a result of a history of in-service failures arising from SCC and chloride-induced SCC. The preliminary options for the SG tubes are in line with current RGP as the materials used for new and replacement SGs; operators in Canada and Germany typically use Alloy 800, whilst operators in France, UK and USA favour Alloy 690TT. Following the outcome of its assessment, the RP has selected Alloy 690TT due to the superior thermal conductivity and corrosion resistance. I am content with the RP's justification for the material choices for the SG tubes.

303. The RP makes a number of statements that Alloy 690TT is "not susceptible" and "almost immune" to PWSCC. I consider that there is insufficient data to fully support this claim for the full 60 year life. Whilst there have been, to date, no reported incidences of PWSCC of Alloy 690TT SSCs in operation (since its first application in 1989) and a number of studies have confirmed the high resistance to PWSCC initiation up to 360 °C, some accelerated ageing studies suggest the potential for PWSCC at long operating times (Ref. 97, Ref. 98). There remains a requirement for ISI throughout the full operating life, and I was content to note that this has been recognised by the RP and is included within the management strategy for material integrity.

304. The RP specifies a number of strict supplementary requirements (in addition to that specified within the American Society of Mechanical Engineers (ASME) SB-163 code) on the specification of Alloy 690TT to be used in UK HPR1000 to further reduce the risk of cracking, which I consider to be a positive inclusion at the GDA stage. These

include, a higher minimum chromium content, non-metallic micro-inclusions of type A to D ≤1.5, and control of trace element levels (such as copper, sulphur, phosphorus etc) to reduce the risk of cracking and improve weldability.

305. The RP has selected an alternative nickel-based alloy, Alloy 750X, for the positioning pins of the clevis inserts of the RVI. The radial support key inserts and clevis inserts together limit the rotations and tangential movement of the lower RVI, and provide a load path for lower support plate horizontal loadings while allowing unrestrained radial and axial thermal growth of the core barrel and lower support plate (Ref. 99). Alloy 750X is a high strength precipitation hardened nickel-based alloy commonly used in fasteners and springs. This alloy is the high strength analogue of Alloy 600. A possible alternative (which is also commonly used in PWRs) is a higher chromium content Alloy 718 (17-21% chromium). Whilst both have shown susceptibility to PWSCC in PWR environments, Alloy 750X has suffered to a greater extent than Alloy 718 (Ref. 100, Ref. 97). During GDA, the RP did not include this component for detailed material selection assessments referenced in the safety case. The RVI has a safety classification SC1; given its significance, I consider this to be an Assessment Finding.

AF-UKHPR1000-0119 – The licensee shall justify the material choices of high strength components which are susceptible to stress corrosion cracking within UK HPR1000.

306. Stainless steels are specified for use in the UK HPR1000 in the CRDM adaptor flange, MCL, PZR and the majority of components in the RVI. Austenitic stainless steels have performed very well in PWR environments for many years now. Of the few failures which have occurred, they were attributed to sensitisation, cold work coupled with chemistry control problems such as chloride surface contamination, or out of specification chemistry in areas such as dead legs (Ref. 100, Ref. 97). The RP has specified low carbon level grades for all the austenitic stainless steel components (carbon levels at ≤0.035%) to reduce the risk of sensitisation by grain boundary chromium depletion. The stainless steels specified for the UK HPR1000 are typical for use in PWRs and should provide adequate corrosion resistance within the primary coolant, provided an appropriate chemistry regime is maintained.

307. The steels in the RVI are also susceptible to IASCC. The RP place controls on the low temperature melting elements (lead, sulphur, tin and mercury) in these materials in addition to general high purity and low non-metal inclusion content, in order to reduce enhanced diffusion in the metal matrix. Steels with a high chromium content are generally more resistant to IASCC, which is reflected in the RP’s selection of austenitic steels. Further controls on the design are claimed to help mitigate IASCC, and in particular, the use of an all welded metal reflector structure which eliminates the use of bolts. I consider the material specification and controls to be adequate to minimise the risk of IASCC within UK HPR1000.

308. Cold worked austenitic and martensitic steels are specified for the RVI guide and alignment pins, and RCP shaft seal housing and the RVI hold down spring respectively. These materials are inherently susceptible to cracking due to increased material hardness, localised phase deformation and intra-granular decomposition of the martensitic matrix. A detailed assessment to justify the materials selected was not provided as part of GDA, and this forms part of Assessment Finding AF-UKHPR1000-0119, described above.

309. The RP specifies Alloy 52/52M/152 as the nickel-based alloy filler metals and 308L/309L (L referring to low carbon levels of ≤0.04% to reduce the risk of sensitisation by grain boundary chromium depletion) as the stainless steel filler metals for welds and clad of the PZR, RPV and SG in the UK HPR1000. The clad acts as a high corrosion resistant barrier to underlying LAS from the primary coolant. The RP provided no justification for the selection of these materials during GDA. However

these are typical clad and weld materials for use in PWR environments; they are compatible with the Alloy 690 and austenitic safe ends (the fitting to transition from a post weld heat treated vessel nozzle), and have shown high resistance to PWSCC. I was content with the material choices (Ref. 97).

310. The RP claims that any cracking of the clad materials is not expected to propagate into LAS. I asked for evidence to support this claim as part of my sampled assessment of the SGs in RQ-UK-HPR1000-1640, and the RP responded that due to the clad ductility, propagation is unlikely (Ref. 3). It also noted that should the clad be compromised so that the underlying LAS is exposed to the primary coolant, the defect tolerance assessment indicates no pressure boundary failure will occur in the 60 year lifetime. This area is assessed as part of the Structural Integrity assessment of UK HPR1000 (Ref. 96).

311. Overall, I consider the majority of material choices made for the UK HPR1000 are adequate except for some of the high strength material choices, the justification for which was not presented during GDA, and should be re-visited during the licencing phase as noted in Assessment Finding AF-UKHPR1000-0119.

Fabrication Controls

312. In addition to alloy selection, a material’s susceptibility to SCC can be minimised by removing the residual material stresses and control of the chemical environment. As such, the finishing stages of fabrication can be used to treat metal surfaces to minimise risk of SCC by control of defects (such as contamination and alloy microstructure) and residual stress. This subsection assesses the RP’s supplementary requirements for stress and defect control of the chosen materials.

313. The RP has not specified the manufacturers that would be used for the different SSCs for UK HPR1000 at the GDA phase, and detailed specifications will need to be produced during the site-specific phases. However, a number of supplementary requirements for the SG tubes have been presented by the RP (Ref. 56, Ref. 55). From a Chemistry perspective, those of interest are specifications which impact corrosion resistance and the source term; my assessment of those which impact the latter is discussed in sub-section 4.2.3.1.

314. The RP notes the following key fabrication controls for the SG tubes. I consider these to be in line with RGP.

- Mature melting technologies and modern tube drawing methods in line with RGP (such as pilger rolling) to ensure uniformity, homogeneity, cleanliness and absence of defects.
- Heat treatments (mill anneal, thermal and stress relief) to control chromium carbide formation and grain size.
- Surface finish to ensure uniformity and absence of imperfections, surface carburisation, decarburisation or nitriding.
- Surface cleanliness checks and controls, and specified limits for detrimental elements used in consumables (such as halides, halogens, sulphur-containing ions and phosphorous).
- Material testing at various stages of the process (including chemical composition, microstructure and evidence of corrosion or IGA).

315. In addition to the controls noted for the SG tubes, the RP stipulates the following general practices which will be in place to mitigate risk of SCC and IGA:

- Solution annealing after forging and bending for stainless steel components to avoid material sensitisation and reduce internal stresses.
- Stress relieving heat treatment after buttering to relieve residual stress.

- Surface treatment to avoid stress concentration induced by surface defects and obtaining residual compressive stress (such as shot blasting and stress relieving).
- Minimisation of cold work and grinding.
- Optimised welding procedures to minimise time at temperatures where sensitisation may occur.
- Control of manufacturing processes to avoid contamination by halogen ions.
- Ultrasonic examination to confirm absence of material defects.

316. In general, full documentation detailing the SSC's specifications has not been developed for GDA (with the exception of the SG tubes and RCP), and this is to be completed as normal business by the licensee. I judge that the controls contained within the documents submitted by the RP demonstrate that it has a good understanding of the principles for fabrication control to reduce the risk of SCC and IGA.

4.2.4.2 Chemistry for Materials Integrity

317. In addition to the important material selections that have been described in the previous section, the RP must ensure that the chemistry of the primary circuit is designed to minimise corrosion, which should also consider the materials of construction. The minimisation of corrosion is one of the main purposes of the primary circuit chemistry regime. It is a topic that I have assessed in detail throughout GDA, and it interfaces with several other assessment sub-topics in this report.

318. In PCSR Chapter 21, the RP identifies the main corrosion mechanisms of concern as types of localised corrosion, either pitting or more importantly SCC and IGA, which were discussed in sub-section 4.2.4.1 (Ref. 5). General corrosion is not considered to be a significant mechanism of corrosion loss due to the performance of the materials that have been chosen.

319. I have assessed the primary circuit chemistry, and how it impacts on corrosion, in detail elsewhere in this report. In terms of corrosion prevention and materials choices, the following paragraphs provide a brief summary of that assessment, and links to relevant parts of this report.

320. The two factors that have a significant influence on localised corrosion are hydrogen and the minimisation of impurities. The addition of hydrogen is included within the primary circuit chemistry regime to remove any remnant oxygen, and to lower the electrochemical potential. This has a significant effect on the minimisation of corrosion.

321. I have assessed the target hydrogen concentration for the UK HPR1000 in detail in sub-section 4.2.7. In summary, I concluded that the RP's chosen operating range ([redacted] – 50 $cm^3 kg^{-1}$, with a target towards the lower end of that range) and operating limits (10 – 50 $cm^3 kg^{-1}$) represented RGP, and therefore meant the risks of corrosion had been reduced SFAIRP. A further corrosion mechanism related to the choice of hydrogen concentration is Low Temperature Crack Propagation (LTCP). LTCP is a form of hydrogen embrittlement that depends on the concentration of hydrogen in the water. I have reviewed the possibility of this mechanism occurring in the UK HPR1000 in sub-section 4.2.10.2 (start-up and shutdown chemistry), and concluded that it is not a concern.

322. Alongside control of pH, the control of impurities in the primary circuit is a fundamental part of the chemistry regime, as control of several deleterious species will help to minimise corrosion and therefore the amount of CPs that enter the primary circuit. The primary circuit materials are susceptible to corrosion mechanisms, such as Intergranular Stress Corrosion Cracking (IGSCC) and IGA, and these mechanisms are directly influenced by impurities in the water. IGSCC or IGA can be initiated and

propagated in the presence of a sufficient concentration of inorganic ions (halides such as chloride and fluoride, and other ions such as sulphate) and dissolved oxygen. High impurity levels have the potential to cause SG tube rupture within one cycle of operation and the control of chemistry remains a primary means of preventing this type of failure.

323. Control of impurities is closely connected to the choices that the RP has made in terms of materials selection, and this is assessed in sub-section 4.2.3.1. The ability of the plant to control the concentration of these species, in particular the RCV [CVCS] is assessed in sub-section 4.2.2.

324. The RP has set limits for the primary circuit to control the common impurities that can promote corrosion, as shown in Table 7 (Ref. 32). Sulphate is treated as a control parameter when the coolant temperature is greater than 120 °C, but as a diagnostic parameter when the coolant temperature is below 120 °C.

Table 7: Proposed Operating Window and Limits for Various Primary Circuit Impurities

Normal Operating Mode	Control Parameter	Operating Window (µg kg^{-1})	Limit (µg kg^{-1})
Reactor in Power Normal Shutdown with SGs Normal Shutdown with RIS [SIS] in RHR Mode	Chloride	<150	<1500
	Sulphate	<150	<1500
	Fluoride	<150	<1500

325. In response to RQ-UKHPR1000-1556, the RP justified why these limits are appropriate and reduce risks SFAIRP, stating that further reductions in impurities would require significant investment in additional water treatment facilities which would not significantly affect the corrosion rates of the primary circuit materials (Ref. 3). These limits also correspond with various international guidelines, and I am content that they are appropriate for the UK HPR1000 (Ref. 75).

326. In addition to the proposed limits, the RP presented OPEX from several sources to demonstrate that these limits were reasonable, and that they were achievable. In addition to some high-level data from international operators, the RP presented some data from CGN's plants in China, to demonstrate that the UK HPR1000 would be capable of achieving these limits. The information that the RP provided was very limited. Further data was supplemented in the 'Topic Report on Impurity Control for the Operation', which provided more data; for a number of CGN plants, the data presented was limited to the number of cycles, the number of individual data values, an average, minimum and maximum value, for each species (Ref. 32). However, as the raw data was not reported, there was no possibility to fully sample and assess it, should I have wanted to include it within my sample. As such, although the data summary forms part of the safety case, I do not consider that the data has been adequately included in the case. This is part of Assessment Finding AF-UKHPR1000-0011.

327. In order to prevent SCC, the RP identifies that the chemical environment is an important factor in the reduction of SCC crack initiation and propagation. The RP's strategy is to limit chloride, fluoride and sulphate (identified previously), and therefore a limit on oxygen is not required. This is because with such low levels of chloride, SCC will not occur, even with a small amount of oxygen (Ref. 32). It is also true that oxygen levels will be very low during normal operation, as hydrogen dosing will ensure any

oxygen is consumed (see sub-section 4.2.7). The RP has adopted limits for hydrogen, which provides assurance that the oxygen concentration will be well controlled. Oxygen concentration is a control parameter during start-up, to ensure that reducing conditions are adopted prior to certain temperatures being exceeded. The RP has also adopted a chloride limit that is consistent with international guidelines (Ref. 75).

328. However, this is a further example of a lack of complete OPEX data in the generic safety case, as only a summary from a few plants was provided; it is not clear from the document how many plants this data was taken from. This means that I am unable to assess whether the data is representative of the wider CGN fleet because of the limited amount of data that has been provided. This is linked to Assessment Finding AF-UKHPR1000-0011.

329. Other corrosion mechanisms such as IGSCC and IGA can be caused by the presence of a range of impurities in the primary circuit, such as oxygen, chloride, fluoride and sulphate. The concentration of these species in the primary circuit must therefore be carefully controlled. The RP has defined the operating limits for these species in the 'Limits and Conditions for Operations' report (Ref. 30). I assessed these operating limits in sub-section 4.2.3.3, and I was satisfied that they would reduce the risk of corrosion SFAIRP.

4.2.4.3 External Corrosion

330. Boric acid is corrosive and there have been a number of instances where boric acid has caused substantial thinning of pressure vessel walls from the outside (Ref. 84). The RP has identified this degradation mechanism as a potential risk to the RPV, PZR, SG primary head and the main studs and nuts of the RCP (Ref. 93, Ref. 89, Ref. 59, Ref. 91). I was content that the key areas of risk have been identified by the RP, and as this aspect is not a fundamental chemistry design issue, it is more appropriately assessed at a later stage. I am content by this approach.

331. I would expect the licensee to implement appropriate controls, including an inspection and maintenance programme, to prevent and detect external corrosion as part of their normal inspection and maintenance procedures. The RP has demonstrated understanding of the importance of ISI as the mitigation strategy for boric acid corrosion of at risk SSCs, which I consider to be a positive inclusion at the GDA stage. This is primarily within the remit of the licensee to develop as part of the plant operating instructions; during my assessment I have not noted any reasons why such a strategy cannot be implemented in UK HPR1000.

4.2.4.4 Strengths

332. Based on the evidence presented during GDA, the main strengths I have identified for the assessment of the primary circuit integrity are:

- Adoption of suitable materials to reduce relevant corrosion risks, with adequate justification provided.
- During normal operations, the claimed chemistry controls should offer sufficient protection to the primary circuit materials to minimise the possibility of integrity damage by corrosion.

333. Overall, I consider that the RP has determined an appropriate prevention and management strategy for SSCs in contact with the primary coolant for the licensee to implement and build upon.

4.2.4.5 Outcomes

334. Based on the assessment of the materials integrity and corrosion control of the SSCs in contact with the primary coolant in the UK HPR1000, described in sub-section 4.2.4 above, I have identified one Assessment Finding which will need to be addressed by the licensee which concerns the justification of the material choices of high strength components which are susceptible to stress corrosion cracking within UK HPR1000 (AF-UKHPR1000-0119). I have also identified a number of areas that are relevant to the Assessment Finding concerning inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011), raised earlier in this report.

4.2.4.6 Conclusions

335. The RP has presented an adequate justification of their approach to materials choices, and adequately justified the interaction between those materials and the primary circuit chemistry regime, to minimise corrosion SFAIRP.

336. Based on the outcome of my assessment of secondary circuit material integrity and corrosion in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

337. I have identified one Assessment Finding as a result of this part of my assessment.

338. The principal SAPs relevant to my assessment of the materials integrity are ECH.1, EAD.1 to EAD.4, EMC.13 and EMC.16.

4.2.5 Fuel Deposit Formation

339. The primary coolant of a PWR contains soluble and particulate CPs, mainly arising from releases from the outer oxide layers of SG tubes and other out of core surfaces. These CPs may deposit on fuel cladding surfaces as fuel deposits during power operations, due to chemical and thermal hydraulic effects in the core, particularly in areas with high rates of sub-cooled boiling at the clad surface.

340. The effects of fuel deposits on nuclear safety vary depending upon the amount that is present. If significant amounts of deposits are present, this can result in damage to the fuel cladding through enhanced corrosion, which occurs due to a combination of reduced heat transfer and enhanced concentration of chemical species (for example those containing lithium) at the clad surface; this phenomenon is known as Crud Induced Localised Corrosion (CILC). In the event that significant fuel deposits occur with boron trapped in the deposit layer, a distorted power profile can occur; this is known as Crud Induced Power Shift (CIPS). If left to develop uncontrolled, CILC would result in undesirable leaks of radioactive species from the fuel and CIPS could lead to a loss of shutdown margin. Additionally, any amount of fuel deposit leads to increased plant radiation levels, as discussed in sub-section 4.2.3, leading to the generation of increased amounts of radioactive waste and creating the potential for increased ORE.

341. ONR therefore expects the generation and accumulation of fuel deposits for a modern PWR design to be minimised SFAIRP, and for a robust quantification, characterisation and justification to be made of the fuel deposits expected. I raised RO-UKHPR1000-0015 during Step 3 of GDA due to a lack of such information in the generic safety case for UK HPR1000 (Ref. 4). In response to the RO, the RP provided an estimation of the expected fuel deposits in UK HPR1000, based on modelling performed with its fuel deposit behaviour analysis code, CAMPSIS. The mass, thickness and the radioactivity concentration of the expected fuel deposits was reported, along with sensitivity studies of key parameters influencing the formation of fuel deposits. My assessment of the

information provided, along with that of the Fuel & Core inspector, is detailed in an assessment note, ONR-NR-AN-21-002 (Ref. 101); the Chemistry aspects are summarised below in sub-sections 4.2.5.1 and 4.2.5.2 of this report and the Fuel and Core aspects are summarised in ONR-NR-AR-20-012 (Ref. 102). I considered all of the matters raised in ONR-NR-AN-21-002 in this part of my assessment and I subsequently raised an Assessment Finding and a number of minor shortfalls, as noted in the following sub-sections.

4.2.5.1 Fuel Deposit Estimates

342. The RP sets out a number of sub-arguments in PCSR Chapter 21 in relation to fuel deposits, including that it considers the accumulation of fuel deposits and the risk of fuel cladding corrosion to be minimised SFAIRP in UK HPR1000 through fuel design and chemistry control (Ref. 5).

343. The RP's report 'Assessment of Fuel Crud for UK HPR1000' presents detailed predictions of total fuel deposit mass, thickness and deposited boron mass distributed around the core of UK HPR1000 (Ref. 103). The method used to provide the predictions involved a fuel deposit behaviour analysis code, called CAMPSIS. CAMPSIS was developed by the RP to produce quantitative predictions of the fuel deposits in UK HPR1000 using thermal hydraulic parameter inputs that are generated using another code, LINDEN, which is a sub-channel thermal hydraulic code. Verification and Validation (V&V) evidence for CAMPSIS was provided in 'CAMPSIS-A CRUD Behaviour Analysis Code: Verification and Validation Report', the assessment of which is covered in sub-section 4.2.5.2 (Ref. 104). An assessment of the validity of the LINDEN code is included in ONR's Fuel & Core Step 4 assessment report (Ref. 102).

344. The overall outcome of the RP's fuel deposit predictions using the CAMPSIS code is that low levels of fuel deposits are predicted for UK HPR1000. The RP's review of OPEX on fuel deposits in operating PWRs supports this conclusion, noting that its expectation is that the fuel deposit level in UK HPR1000 will be lighter than most US plants and similar to, or slightly lower than French, German and CPR1000 plants reviewed, with the coolant chemistry and core design presented in GDA (Ref. 103).

345. In line with SAP AV.6, which states: "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.", the report 'Assessment of Fuel Crud for UK HPR1000' presents a series of sensitivity studies on the effect of key chemistry parameters and material corrosion rates on the fuel deposit predictions (Ref. 103). The water chemistry studies were aimed at identifying the key parameters influencing the formation of fuel deposits; primary coolant pH, dissolved hydrogen concentration and lithium-boron concentration were considered. Studies were also performed to consider the sensitivity of the fuel deposit estimates to assumptions made on corrosion release rates in UK HPR1000.

346. Corrosion release input data for different cycles were determined from the RP's own experiments using Alloy 690TT, 304 stainless steel and 316L stainless steel test specimens. The experiments were carried out at a single dissolved hydrogen concentration; however, a sensitivity analysis was performed to vary the modelled hydrogen concentration and consider the effect of an increasingly reducing environment on corrosion release rate. The RP notes that even at the lower end of the hydrogen concentration range studied, reducing conditions (which depress corrosion) are attained and, therefore, increasing the hydrogen concentration has little effect on the mass of fuel deposits produced. The RP uses OPEX from three Ringhals PWR plants and several CPR1000 units, where the hydrogen concentration was raised with negligible impact on corrosion behaviour, to support its conclusion that the results of its fuel deposit modelling are not highly sensitive to the hydrogen concentration. This is in

contrast to other areas of the generic safety case, where the RP argues that hydrogen concentration is important in the reduction of corrosion. The evidence used to perform the hydrogen sensitivity study is from a narrow base, and I consider that the safety case should be strengthened with further evidence on the role that hydrogen plays in fuel deposit formation. I consider this to be an Assessment Finding.

AF-UKHPR1000-0120 – The licensee shall demonstrate the impact of the primary circuit hydrogen concentration on fuel deposit formation and behaviour in UK HPR1000, and justify that the proposed primary circuit hydrogen operating range reduces relevant risks so far as is reasonably practicable.

347. The effect of pH_T variation is also studied between 7.0 and 7.4. As the corrosion release experiment was performed at a single pH, an adjustment factor is applied based on EPRI CPs research data. The RP's analysis shows that an elevated pH results in a small decrease in core fuel deposit mass, thickness and boron deposition mass. OPEX from Ringhals, where pH_T was elevated to 7.4 in three reactors with resultant decreases in corrosion release rates, is used to support this result. Whilst the result of the analysis points to a small positive effect of increased pH on fuel deposit levels, I am content that the RP has reached an optimised position overall with regard to primary circuit pH control, as described in sub-section 4.2.3.3 of this report.

348. The RP also presents an analysis of the sensitivity of the fuel deposit estimates to bulk boron concentration in order to compare the use of natural boric acid (used in CPR1000) and EBA. The results of the analysis show generally less boron deposition mass throughout the cycle with EBA compared to natural boric acid, except for at the end of a cycle; this is because when most deposited boron returns to the bulk coolant a slightly larger ^{10}B deposition mass is predicted with EBA due to its higher enrichment. Overall, the results support the RP's decision to employ EBA in UK HPR1000.

349. The RP's analysis of the sensitivity of its fuel deposit estimates to corrosion release rate shows that core fuel deposit mass, thickness and boron deposition mass are directly proportionate to corrosion release rate. Since the corrosion release rates for UK HPR1000 are estimated from experimental data and the fuel deposit estimate was shown to be sensitive to corrosion release rate, the RP repeated the UK HPR1000 fuel deposit estimates using corrosion release rate data from operating CPR1000 units. With CPR1000 corrosion data the maximum fuel deposit thickness of UK HPR1000 increases slightly (from [redacted] microns to [redacted] microns), however the RP notes that the conclusion of the report 'Assessment of Fuel Crud for UK HPR1000', that the maximum fuel deposit thickness for UK HPR1000 is less than [redacted] μm based on current core design, material choice, and chemical control, is not challenged (Ref. 103).

350. The UK HPR1000 will dose zinc (in the form of DZA) into the primary circuit, to reduce corrosion and doses to operators (from the resultant reduction in CP generation). The RP has modelled the effect that zinc dosing will have on fuel deposit formation, in; my assessment of these aspects is detailed in sub-section 4.2.6.2 (Ref. 103).

351. I am content that the RP has appropriately considered the sensitivity of its fuel deposit estimates to the key chemistry and material corrosion input parameters, in line with SAP AV.6, and has demonstrated that its estimates are not highly sensitive to variation in these parameters (Ref. 2). Assessment of the sensitivity of the estimates to thermal hydraulic input parameters was performed by the Fuel and Core inspector and recorded in (Ref. 102).

352. 'Assessment of Fuel Crud for UK HPR1000' also presents some OPEX data from the CPR1000 fleet in China and, by also generating fuel deposit predictions for a CPR1000 plant using the UK HPR1000 methods, aims to provide some additional confidence in the conclusions drawn (Ref. 103). The RP's analysis shows that the deposited ^{10}B mass is predicted to be lower for UK HPR1000 than CPR1000 plants

and presents evidence to show that CPR1000 plants have not experienced problems due to CIPS. The conclusion of the OPEX review also implies that the CILC phenomenon will not occur in the UK HPR1000 plant because only 'light' deposits are expected to be observed (Ref. 103). Assessment of the consequences of predicted fuel deposits for UK HPR1000, including the likelihood of occurrences of CILC and CIPS and the impact of deposition on the consequences of individual faults, was performed by the Fuel and Core inspector (Ref. 102).

353. The outcome of the RP's modelling is that low levels of fuel deposits are expected for UK HPR1000; this is supported by a range of OPEX suggesting that the level of deposits in UK HPR1000 will be lighter than CPR1000 and other PWR plants reviewed, with the coolant chemistry and core design presented in GDA. Overall, from a Chemistry perspective I am satisfied that the RP has developed quantitative estimates of fuel deposits for UK HPR1000 and used these, together with sensitivity analyses and international OPEX data, to provide a robust assessment of the range of fuel deposits that could occur for UK HPR1000.

4.2.5.2 CAMPSIS Verification and Validation

354. As explained in ONR's guidance on the validation of computer codes and calculation methods (NS-TAST-GD-042), ONR expects a dutyholder to present validation for the computer codes/calculation methods used in a safety case (Ref. 6). Validation should provide the evidence that the computer code or calculation method is fit for purpose by comparison of the RP's results with data from experiments or other trusted sources. In addition, a verification process should demonstrate that the model specification has been complied with and that controlling physical equations have been correctly translated into the computer code. My assessment of the RP's V&V evidence for CAMPSIS against the expectations set out in NS-TAST-GD-042 is described below.

355. As part of its response to RO-UKHPR1000-0015, the RP produced a report detailing the V&V of the CAMPSIS code, which presents a combination of separate effect tests, sensitivity studies on key chemical and thermal hydraulic parameters and integral effect tests using plant data (Ref. 4, Ref. 104). A range of data is used in comparisons with CAMPSIS-simulated results, including coolant activity data from operating CPR1000 units and from Ringhals Unit C PWR in Sweden, publicly available data on the effect of zinc injection on coolant activity, and experimental data (both publicly available and the RP's own) on corrosion release rates.

356. I note that the majority of the validation with plant data relies on comparisons of coolant activities with the equivalent CAMPSIS simulations. The RP explained in response to RQ-UKHPR1000-1509 that other plant data, such as shutdown nickel release and the nickel-iron-chromium ratio, is not regularly measured in CPR1000 plants and therefore limited data of this type were available to the RP for use in the validation work (Ref. 3). The RP claims that whilst a direct correlation cannot be made between coolant activities and fuel deposits, the use of coolant activity data for the majority of the validation work is appropriate, as fuel deposits are a key contributor to coolant activities. Whilst I considered this to be an acceptable position for the purposes of resolving the RO during GDA, I judge that further development of the modelling is required in future to strengthen the RP's case for the validity of its fuel deposit estimates (this is a minor shortfall, as recorded in paragraph 358).

357. The comparisons of CAMPSIS-simulated coolant activities (and dose rates and shutdown nickel release rates where these were available) with CPR1000 and Ringhals' plant data presented in the V&V report appear to correlate sufficiently well to support the RP's claims on the validity of the model. As part of the separate effects tests, the RP also presented comparisons between CAMPSIS-simulated results for nickel and iron solubility and experimental results of the same from literature sources; of these some comparisons provide a close correlation, whereas others are less well

correlated. In response to RQ-UKHPR1000-1509, the RP reasoned that the impact of the gap between the simulated and measured solubility data on the overall fuel deposit estimates is minimal, since the impact of solubility on deposits is small compared to other factors, such as the core boiling area (Ref. 3).

358. Whilst the validation evidence for CAMPSIS could have been strengthened with the use of additional operating plant data, I consider that the evidence presented is sufficient to give confidence in the validity of the RP’s fuel deposit estimates for UK HPR1000 for GDA. However, in line with the expectation set out in SAPs AV.1 and AV.2, I consider that the fuel deposit estimates for UK HPR1000 should continue to be developed and refined by the licensee, such as to take account of more directly relevant plant data and test results (for example scrape data), including from other HPR1000 plants as this information becomes available (Ref. 2). I consider this to be a minor shortfall.

359. During GDA, the RP was not able to produce a comparison of the fuel deposit estimate from CAMPSIS with another estimate from an independent code. As noted in NS-TAST-GD-042, useful information can be obtained by comparing one calculation method with another diverse method (Ref. 6). I consider that the presentation of a code-to-code comparison would strengthen the RP’s case in support of the validity of its fuel deposit estimates; I judge this to be a minor shortfall.

360. Regarding the verification of CAMPSIS, the RP notes in its V&V report that the correctness of the coding and programming has been checked by manual calculations and the depletion model verified using another code, PALM. The appropriateness and adequacy of the PALM code has been assessed in the Severe Accident topic (Ref. 105). Whilst this approach to verification appears reasonable, no specific evidence of these activities was provided in the report, nor in response to RQ-UKHPR1000-1509 (Ref. 3). The RP does claim in response to the RQ, that the water chemistry sensitivity analysis presented in ‘Assessment of Fuel Crud for UK HPR1000’ contributes to the code verification by confirming that deposit estimates are reasonable within a given range of input parameters, such as pH and corrosion release rate (Ref. 103). Overall, I consider that further evidence to support the RP’s claims regarding the verification of CAMPSIS would strengthen the safety case; I therefore consider this to be a minor shortfall.

361. Chapter 3 of ‘CAMPSIS-A CRUD Behaviour Analysis Code: Verification and Validation Report’ presents the underlying theory for each of the models within CAMPSIS and the derivation of the equations used to model the various processes involved in fuel deposition (Ref. 104). Whilst the information presented appears reasonable, it should be noted that my assessment scope did not include a ‘deep dive’ review of the theory underlying the CAMPSIS code.

362. In addition to the justification of the modelling process, NS-TAST-GD-042 notes that there is a need to establish that the code correctly represents the physical model by ensuring that a systematic approach has been adopted for designing, coding, testing and documenting the computer programme. I am content that the quality assurance information provided in response to RQ-UKHPR1000-1509 satisfies this expectation to the extent required for GDA (Ref. 3).

4.2.5.3 Effects of Coolant Chemistry on Fuel Cladding

363. The chemistry of the primary circuit coolant can have an impact on the fuel cladding, and several species should be carefully controlled to prevent any deleterious impact on one of the main barriers to the transport of radioactivity within the plant. The following section briefly discusses these species, and the actions and mitigations that have been proposed by the RP.

364. The fuel which is proposed to be used in the UK HPR1000 is clad in the M5 material (a Zircaloy) and is manufactured by Framatome. The RP reports that the fuel suppliers have specified that the lithium hydroxide concentration in the primary circuit should be less than 4 mg kg^{-1}, due to concerns that at higher lithium hydroxide concentrations, the cladding material is susceptible to corrosion. International guidance also supports a higher lithium hydroxide limit of this magnitude (Ref. 75). To provide some margin to the manufacturer's limit, the RP has proposed an operating limit of 3.5 mg kg^{-1} (Ref. 76). The choice of lithium hydroxide concentration is a balance between achieving the highest possible pH for a given concentration of boric acid, whilst minimising the risk of fuel cladding corrosion. I have assessed the impact on pH in sub-section 4.2.3, and I am content that the regime specified is adequate. Therefore, I am content with the lithium hydroxide concentration limit described by the RP.

365. Hydrogen is added to the primary circuit to suppress radiolysis, and to minimise corrosion of stainless steels and nickel-based alloys. However, high concentrations (greater than 50 cm^3 kg^{-1}) of hydrogen are known to embrittle the zirconium oxide that does form on the surface of the fuel cladding material, and therefore an upper hydrogen limit is necessary to prevent this mechanism from occurring. The RP has set an upper limit on the hydrogen concentration in the primary circuit of 50 cm^3 kg^{-1}, which I assess in sub-section 4.2.7.6 in more detail, but I am content that this limit should prevent any risk of embrittlement of the fuel cladding.

366. The RP identifies fluoride and chloride as species that pose a particular risk of causing corrosion of the fuel clad, and places limits on them to reduce this risk as much as possible. The RP has identified a balance between limiting the concentration of these species as much as possible to reduce the risks of corrosion (through clean-up of the coolant in the RCV [CVCS]), and the cost of performing this purification. I assessed the RP's approach to the minimisation of impurities and the limits that have been defined in sub-section 4.2.3.3, and I was satisfied with the proposals defined by the RP.

4.2.5.4 Strengths

367. Whilst I have identified several minor shortfalls in this part of the case, I consider that, overall, the RP has provided a robust assessment of the range of fuel deposits that could occur for UK HPR1000, based on the information available at GDA. In doing so, I consider that the RP has met the intent of RO-UKHPR1000-0015, SAP ECH.1 and relevant aspects of NS-TAST-GD-088 (Ref. 4, Ref. 2, Ref. 6).

4.2.5.5 Outcomes

368. Based upon the assessment of the fuel deposit estimates for UK HPR1000 described in sub-sections 4.2.5.1 and 4.2.5.2 above, and in line with SAP AV.6, I am satisfied that the RP has developed quantitative estimates of fuel deposits for UK HPR1000 and used these, together with sensitivity analyses and international OPEX data, to provide a robust assessment of the range of fuel deposits that could occur for UK HPR1000 (Ref. 2). The outcome of the RP's modelling is that low levels of fuel deposits are expected for UK HPR1000; this is supported by a range of OPEX suggesting that the level of deposits in UK HPR1000 will be lighter than CPR1000 and other PWR plants reviewed, with the coolant chemistry and core design presented in GDA. Overall, I consider that appropriate supporting evidence has been presented to support the RP's arguments that the accumulation of fuel deposits and the risk of fuel cladding corrosion have been minimised SFAIRP based on the chosen chemistry for UK HPR1000.

369. I have identified one Assessment Finding, which needs to be addressed by the licensee, concerning the demonstration of the impact of the primary circuit hydrogen concentration on fuel deposit formation (AF-UKHPR1000-0120). I also identified several minor shortfalls as discussed in sub-sections 4.2.5.1 and 4.2.5.2 above.

4.2.5.6 Conclusions

370. Based on the outcome of my assessment of the fuel deposit estimates and the impact of primary coolant chemistry on fuel deposits for UK HPR1000, I have concluded that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

4.2.6 Zinc Addition

371. As part of the aims of the overall primary circuit chemistry regime to reduce corrosion and to minimise radiation doses, the RP is proposing to inject zinc, in the form of DZA, into the primary circuit of the UK HPR1000. Zinc has become an increasingly common additive to PWR primary circuits worldwide, typically at concentrations of up to 20 µg kg^{-1}. Both PWR designs to complete GDA in the last decade have included zinc dosing. During Step 3 of GDA, zinc injection was not included as part of the reference design of the UK HPR1000. However, through Step 3, the RP firstly took the position that zinc injection represented good practice, and then conducted a feasibility study to assess whether zinc injection was reasonably practicable for UK HPR1000 (Ref. 106). Subsequently, during Step 4 of GDA, a Category 3 modification was made to incorporate zinc injection into the reference design (Ref. 107).

372. The RP is therefore proposing to inject zinc in the form of DZA (depleted in ^{64}Zn to minimise the production of ^{65}Zn) into the primary circuit, via injection equipment that delivers zinc into the RCV [CVCS], to give a concentration in the primary circuit of between [redacted] and [redacted] µg kg^{-1}.

373. In addition to the benefits highlighted above, zinc injection presents some possible detriments. These include the formation of zinc deposits on the fuel, consumption of the ion exchange beds and the influence of acetate on the fate of ^{14}C. I assess both the possible benefits and the detriments in the following sections.

374. The RP has included three arguments relevant to the inclusion of zinc injection within the generic design of UK HPR1000 that support the claim "3.3.10 – the chemistry aspects of the plant design have been developed to reduce the nuclear safety risk SFAIRP", which state that zinc will minimise primary circuit corrosion, the generation of radionuclides, and minimise leakage of the RCP [RCS] (Ref. 5).

375. I assess each of these arguments in the following sections of this report. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment, and provide guidance on the use of zinc in civil nuclear plants (Ref. 6).

376. The RP has provided two documents which seek to justify the use of zinc in the UK HPR1000, and these provide a detailed explanation for the mechanism by which zinc injection reduces corrosion and dose rates (Ref. 106, Ref. 108). In summary, zinc has been shown to become incorporated into the oxide spinel of stainless steels and nickel-based alloys, changing the morphology of the oxide, and thereby increasing the corrosion resistance of the material. The RP describes how studies have shown that on previously un-oxidised stainless steel and nickel-based alloys, the presence of zinc produces a significantly thinner oxide layer on the surface, with a much smaller grain size, which results in enhanced corrosion properties, both in terms of general corrosion, and stress corrosion cracking (Ref. 108).

377. In terms of dose reduction, zinc competes with cobalt within the oxide structure of the steel or nickel-based alloy. This prevents cobalt from residing within the oxide,

resulting in less cobalt being retained and activated in the structural components. This therefore reduces the dose caused due to ^{60}Co, which is one of the main radionuclides responsible for operator doses in a PWR.

378. During Step 3 of GDA, I employed a TSC to conduct a review of current best practice in PWR operation, which included the topic of zinc injection (Ref. 109). I have used their review in my assessment that follows.

4.2.6.1 Potential Benefits

Impact on Dose Rates

379. The RP has used OPEX from a number of operating plants as evidence to support the argument that the use of zinc will reduce the risk of radiation dose to workers. The RP selected the plants based upon a set of criteria which were chosen to match closely the design and materials of construction of the UK HPR1000. I note that this OPEX is only taken from a small number of plants, and that only one plant (Tomari 3) has Alloy 690TT SGs.

380. The source of most of the data presented is from plants that have been operating for a considerable time and, following the introduction of zinc dosing, have replaced SGs. The replacement of SGs means that a significant area of new metal is exposed to the primary circuit; the effect of zinc following this change presents good evidence regarding the effectiveness of the change of chemistry.

381. The evidence provided by the RP is consistent in showing a significant effect in terms of dose rate reduction in plants that have begun to dose zinc. Data from a plant that began operating with zinc dosing also shows positive results. The decision about whether to start zinc dosing during HFT is assessed further in sub-section 4.2.11.

382. The safety case is largely reliant on publicly available OPEX, much of which is at least ten years old. The RP does not have any experience of operating a plant in China with zinc, and therefore no data can be presented from its own operations. The inclusion of some data from EDF-operated French plants provides further support to the arguments about the effectiveness of zinc. Zinc dosing is further supported in a recent review of worldwide data, which show a consistent reduction in dose rates for plants that have either adopted zinc dosing midway through their operating life, or from the first cycle (Ref. 75). I am therefore content that the RP has adequately demonstrated that injection of zinc into the primary circuit of the UK HPR1000 will likely have a benefit in terms of dose reduction.

Impact on Corrosion

383. The RP claims that a reduction in corrosion of the stainless steel and nickel-based alloys is to be expected when zinc is injected into the primary circuit of the UK HPR1000, and presents various pieces of evidence to support this (Ref. 106). The RP produced a new report that provided a more detailed analysis of the impact that zinc injection would have on the corrosion resistance of primary circuit materials (Ref. 108). For both stainless steels and nickel-based alloys, the report considers the effect of zinc on the structure of the oxide film, the mechanism by which zinc influences the corrosion resistance of the material, and the effects that zinc has on the corrosion behaviour of the material. The structure is logical, and it allows for a clear presentation of the evidence.

384. In summary, the RP's evidence shows that:

- Zinc is readily incorporated into the oxide layer, forming a stable insoluble species.

- Zinc dosing improves the corrosion performance of the primary circuit materials, reducing both general corrosion and SCC.
- Much of the evidence presented is in experiments where 50 μg kg^{-1} zinc has been used, and some of the evidence suggests the performance of the materials increase as the concentration increases to 50 μg kg^{-1}. This is a significantly higher concentration than is proposed for the UK HPR1000 during normal operations (between [redacted] and [redacted] μg kg^{-1}). The RP has proposed to dose zinc at 50 μg kg^{-1} during commissioning, and it is during these processes (where the oxide layer will first form) where a higher zinc concentration is likely to have the greatest effect. Once the oxide has formed, the evidence suggests that a lower normal operating zinc concentration will not have a deleterious effect.

385. In a similar way to the dose rate review, the RP is largely reliant on publicly available evidence to support the claims that are being made about the introduction of zinc dosing. However, a consistent picture is presented, and this largely agrees with other independent reviews of the effectiveness of zinc dosing (Ref. 75). I note that the RP does not refer to the corrosion data that was generated in support of the fuel deposits assessment model, CAMPSIS, which I describe in more detail in sub-sections 4.2.3 and 4.2.5, but does provide some evidence that zinc will reduce corrosion rates in stainless steels and Alloy 690TT.

386. Intuitively, to achieve a reduction in operator dose (which the literature supports) a reduction in corrosion is also highly plausible. Although I am content that it is likely that zinc will have a positive effect in terms of the corrosion performance of stainless steels and Alloy 690, I consider the evidence presented by the RP in support of this claim to be limited, and further plant-specific evidence should be generated by the licensee to support the claims made about zinc injection. I therefore consider this to be a minor shortfall.

387. Although the RP has not claimed a specific quantified benefit from zinc addition, from either dose reduction or a reduction in corrosion, it has provided sufficient qualitative data to argue that zinc will yield significant benefits in terms of both of these aspects. However, I judge that the safety case would be improved if these effects were quantified, and I therefore consider this to be a minor shortfall.

4.2.6.2 Potential Detriments

388. In addition to the benefits described above, there are several potential detriments in using zinc in the primary circuits of PWRs, and these are described in the following sub-sections. During Step 3 of GDA, the RP presented high level arguments about the potential detriments resulting from zinc injection. During Step 4 of GDA, much more detail was provided.

389. The RP has chosen to use DZA, rather than natural zinc. DZA is depleted in ^{64}Zn (to less than 1 at%), as this radionuclide can become activated to ^{65}Zn, which emits high energy gamma rays.

Zinc Precipitation and Effects of Fuel Deposits

390. The RP identifies three ways in which zinc addition could affect the performance of the fuel cladding:

- Zinc addition could affect the performance of the fuel cladding through precipitation on fuel rod surfaces.
- Zinc addition could affect the corrosion and release of CPs from out-of-core surfaces, changing the mass of deposit that accumulates on the fuel.
- Zinc addition could lead to zinc precipitation in the core.

391. The RP's arguments about the effect of zinc dosing on fuel cladding rely on OPEX from plants that have operated with zinc over the last few decades. As I stated previously, Chinese plants do not normally operate with zinc in the primary circuit, and so the use of worldwide OPEX is a reasonable approach.

392. The RP reports results from various plants which had begun to dose zinc, where detailed visual inspections and oxide measurements were conducted to determine if there had been any acceleration of cladding corrosion due to zinc addition (Ref. 106). These results, presented at international conferences, showed that no issues with cladding corrosion were observed, and oxide thicknesses were identical to pre-exposure predictions. These results were taken from a variety of fuel cladding materials, but I note that they also include results from fuel clad in M5, which is the fuel cladding to be used in the UK HPR1000. I am content with these arguments.

393. The RP states that zinc has the following effect on CP and activated CP transport:

- Zinc will reduce the corrosion rate of out-of-core surfaces if present from the initiation of corrosion, which means fewer CPs will be released into the coolant, and therefore the total mass of deposits that accumulate on the fuel could be reduced, decreasing CIPS risk.
- Zinc will increase the concentration of activated CPs in the coolant by preventing them from depositing on out-of-core surfaces.
- Zinc could also deposit on the fuel, increasing CIPS risk. If the coolant concentration of CPs remained constant, more boiling will result in more deposition of CPs on the core. It is expected that there will be an increase in the fraction of zinc injected that is deposited in the core. Most zinc programmes control zinc concentration in the coolant. Therefore, for a higher duty core, the rate of injection will be greater and therefore the mass of zinc deposited on the core will be higher.

394. The RP summarises these effects by saying that if zinc is applied during HFT, then the benefits of the reduced corrosion rate will more than compensate for the effects of increased activated CP concentration. I will assess the potential to dose zinc during HFT in sub-section 4.2.11.

395. Although the review that the RP has described uses data from more than ten years ago, I note that the RP also states that since that review, no other plants have reported CILC or CIPS being caused by zinc injection. The RP reports that the Westinghouse modelling work has been used to predict if a plant will suffer from zinc deposition within the core, by plotting core boiling duty against zinc concentration. Generally, a high zinc concentration (at least 50 $\mu g\ kg^{-1}$) and a low boiling core duty indicates zinc deposition on fuel. The UK HPR1000 has a relatively high core boiling duty, and the RP will adopt a lower zinc concentration (around 10 $\mu g\ kg^{-1}$).

396. In response to RO-UKHPR1000-0015 which was raised to analyse the propensity to form fuel deposits in the UK HPR1000, the RP has conducted modelling work to estimate the likely quantities of deposits. Part of this modelling has accommodated the introduction of zinc injection. I have assessed this aspect of the safety case in sub-section 4.2.5 of this report. However, in the context of my assessment of zinc addition, the RP conducted some tests to analyse the effect that zinc had on corrosion rates of Stainless Steel and Alloy 690TT, which they used to consider the effect on fuel deposit formation. Although limited in nature, the results showed that the introduction of zinc resulted in reduced corrosion, together with a reduction in the amount of fuel deposits that were produced, although the evidence was quite limited. Whilst this appears to be a reasonable assumption, the RP could not provide any direct evidence to link zinc with a reduction in fuel deposits. In my assessment of the closure of RO-UKHPR1000-0015, I concluded that further evidence should be generated to support any future

claims that zinc injection will reduce fuel deposit formation (Ref. 110). I therefore consider this to be a minor shortfall.

397. The formation of fuel deposits on nuclear fuel can be tolerated if the deposit is porous. This allows boiling of the coolant within the pores of the deposit, which is an efficient heat-removal mechanism. Species like silica form a denser deposit that prevents this form of heat transfer, and leads to higher fuel temperatures. This could result in fuel damage (CILC), if not controlled. Zinc silicate is particularly insoluble, and some operators have chosen to provide a limit on silica when adopting zinc injection. In the response to RQ-UKHPR1000-0701, the RP stated that the amount of silica co-precipitate would be less than other oxide species, such as zinc oxide (ZnO) and zinc ferrite (of the general formula $Zn_xFe_{3-x}O_4$) (Ref. 3).

398. Silica is difficult to remove from the circuit once present, and bleed and feed procedures are normally required to decrease the concentration of this species. The RP has decided to categorise silica as a control parameter during periods of operating with zinc injection due to concerns about zeolite formation, with a limit of [redacted] mg kg^{-1}, and an expected value of [redacted] mg kg^{-1} (Ref. 32). Should the limit be exceeded, the RP states that zinc injection should cease, to minimise the risk to the fuel. The limit (of [redacted] mg kg^{-1}) is consistent with limits identified previously, and I am content that it is adequate (Ref. 111).

399. I also asked about the need for limits on other species in the primary circuit, as a consequence of the decision to implement zinc injection in RQ-UKHPR1000-0702 (Ref. 3). The RP responded that the control of nickel is also an important factor in the consequences of fuel deposit formation – nickel is the main element within fuel deposits. During zinc injection, there may be perturbation of the nickel concentration within the primary circuit, which the RP has indicated should be a control parameter. The fuel vendor has also stated that nickel concentration should be controlled and has specified a limit of [redacted] µg kg^{-1}, which has been adopted by the RP. As discussed earlier in this section, zinc is preferentially bonded within the oxide layer on the surface of steels and nickel-based alloys, releasing some nickel into the coolant. As discussed in sub-section 4.2.11, the RP is recommending that zinc injection be first employed during HFT, which means that fuel will not be present when any significant changes to the morphology of the oxide will be first encountered. The evidence also supports the view that once the oxide layer has been formed, subsequent changes will be less significant. Therefore, I expect that any nickel production will be minimal during normal operation. However, I am also content that it is appropriate to adopt the suggested limit.

400. In conclusion, I am content that the RP has adequately described the risks associated with zinc injection and fuel deposits, and that they have been reduced SFAIRP.

Effects of Acetate

401. In order to dose a reactor accurately to levels of several µg kg^{-1}, a solution of zinc is normally added in the form of zinc acetate, as this is readily soluble. The acetate is an additional source of carbon to the reactor, and the addition of natural carbon (^{12}C) could affect the distribution of radioactive ^{14}C (^{13}C in zinc acetate will be activated and generates ^{14}C by reaction ^{13}C (n, γ) ^{14}C).

402. The RP claims several factors limit the generation of ^{14}C from the carbon in acetate:

- Carbon species in the reactor coolant are volatile. Therefore, there is little residence time in the core.
- The neutron capture cross-section of ^{13}C is small especially compared to that of ^{14}N or ^{17}O, which are the significant sources of ^{14}C generation in PWRs.

- The concentration of ^{13}C is very low (when adopting [redacted] µg kg^{-1} of zinc in the coolant).

403. In response to RQ-UKHPR1000-0701, the RP provided some estimates of how much of an increase in ^{14}C would arise from the use of zinc acetate (Ref. 3). The RP reported that in comparison to the normal generation of ^{14}C from ^{17}O in the coolant, the ^{14}C due to zinc acetate is at least eight orders of magnitude less. It is possible that the ^{14}C that arises from acetate may affect the distribution between gaseous and liquid discharges. The RP has not explicitly considered this as part of their assessment of the impact of zinc injection, and I therefore consider this to be a minor shortfall.

Generation of Zinc-65

404. Naturally occurring zinc contains 49% ^{64}Zn, which can be activated in the core to form ^{65}Zn. This isotope decays by internal capture and positron emission, and it can therefore be difficult to detect. The use of natural zinc at some reactors has caused an increase in short-term radiation. To avoid these issues, most operators use zinc that has been depleted in ^{64}Zn. The RP will also use DZA (<1% ^{64}Zn), and given the low concentration of zinc that is proposed to be dosed into the primary circuit (<[redacted] µg kg^{-1}), the concentration of ^{65}Zn will be very low (Ref. 106). The RP also indicated in the response to RQ-UKHPR1000-0701 that in older plants that used natural zinc, the concentration was equivalent to 0.37 GBq t^{-1}; depleted zinc decreases the ^{64}Zn abundance from 48.6% to less than 1% (Ref. 3). I am therefore content that the proposal to use depleted zinc reduces risks SFAIRP.

Effects on Purification Systems and Wastes

405. Within the RCV [CVCS] ion exchange resins, zinc is preferentially absorbed onto the mixed/cation beds, forcing lithium from the resins, and potentially reducing the capacity of the resins for radioactive species.

406. Due to Chinese plants operating without zinc injection, the RP has used an OPEX review to provide information about the likely impact on the purification systems. They claim that although zinc will selectively absorb onto the resins in the RCV [CVCS], the concentration of zinc in the coolant is so low that the impact on resins is limited. The RP reports an EPRI study, which reviewed the operation of a series of mostly US plants following the introduction of zinc dosing. This review shows that none of the roughly twenty plants reported a change in terms of resin performance. Similar results were reported in terms of filter performance (Ref. 112).

407. The RP has provided an assessment of the likely impact on resin performance. Assuming a zinc injection rate of [redacted] g d^{-1}, a cycle would result in [redacted] kg of zinc entering the primary circuit. The RP estimates the impact this would have on the resins compared to the expected requirement for nickel uptake on the resins. With a total mixed bed capacity of [redacted] equivalents, the analysis shows that a nickel release during shutdown ([redacted] equivalents), normal nickel removal during the cycle ([redacted] equivalents), and zinc ([redacted] equivalents), would account for less than half of one mixed bed (Ref. 106). The RP also states that through the use of zinc, corrosion of the primary circuit materials is expected to reduce, which would also have a beneficial effect on the resin performance, and hence the quantity of waste produced.

408. The RP also describes the effect on other waste generation. In terms of gaseous waste, the RP reports that it has not found any OPEX to suggest that zinc injection significantly affects gaseous waste generation. The increased production of ^{14}C could lead to greater discharges. In response to RQ-UKHPR1000-0488, the RP estimated that significantly less than 1% of the total ^{14}C would be due to acetate (Ref. 3). As I have previously mentioned, the RP has not considered in detail the potential

distribution of ^{14}C between gaseous and liquid discharges, which I consider to be a minor shortfall. The TEU [LWTS] is assessed in more detail in sub-section 4.4.3.1.

4.2.6.3 Other Considerations

409. There are a number of other factors that may result from zinc dosing:

- The RP has provided a review of the impact of zinc dosing on other systems that are connected to the primary circuit, in the response to RQ-UKHPR1000-0702 (Ref. 3). It has concluded that no impact is expected for those systems, such as the RCV [CVCS], TEP [CSTS] and the REN [NSS]. The RP confirmed that zinc will not get into contact with the IRWST and BASTs.
- In RQ-UKHPR1000-0702, I asked if the introduction of zinc would deleteriously affect the measurement of other species by the REN [NSS] (Ref. 3). The review conducted by the RP indicated that no issues were expected for the typical means of measuring the relevant species; the details of specific measurement instruments are not required to be specified during GDA. Other sampling requirements associated with zinc injection are assessed further in sub-section 4.2.8.1.
- In response to RQ-UKHPR1000-0701, the RP confirmed that despite the topic report stating that an evaluation of tank sizing would not be completed until the site-specific stages, layout requirements formed part of the modification (Ref. 3, Ref. 107). Although the design is not fully developed, a feasibility study has been conducted by the layout designer, which confirmed there was sufficient space for the dosing tank and associated equipment to be housed adequately within the relevant building; development of the modification is normal business. Zinc injection is not novel, nor is the system dissimilar from other existing plants; following the information provided about the location of the dosing tank provided by the RP, I am content that the RP has provided sufficient information at this stage of the development of the design.

4.2.6.4 Strengths

410. Despite limited access to plant data, the RP has produced a coherent justification for the benefits of zinc addition, using a thorough review of OPEX that is available in the literature. Although the RP has not claimed a specific quantified benefit from zinc addition, it has provided sufficient qualitative data to argue that zinc will yield significant benefits in terms of corrosion minimisation and ORE. This has been balanced by an adequate justification that any detriments from zinc addition are likely to be limited.

4.2.6.5 Outcomes

411. Based upon the assessment of zinc addition in UK HPR1000 described in sub-section 4.2.6 above, I have not identified any Assessment Findings.

412. I identified five minor shortfalls as discussed in sub-section 4.2.6 above.

4.2.6.6 Conclusions

413. The RP has adequately justified that the inclusion of zinc injection will be beneficial in terms of corrosion minimisation, and a reduction in operator doses. Any potential detriments have been shown to be not significant.

414. Based on the outcome of my assessment of zinc addition in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

415. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment, which discusses the use of zinc (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the safety case.

4.2.7 Hydrogen Dosing

416. All PWRs in operation today dose their primary coolant with hydrogen gas to control corrosion and radiolysis. There are two reasons for this:

- to suppress any oxidising species that are produced in the core; and
- to provide a low electrochemical potential to minimise the likelihood of corrosion. This also acts to limit out of core radiation fields.

417. The concentration of hydrogen in the primary circuit is the result of a combination of the amount of hydrogen added via the dosing system, that generated in the circuit due to corrosion, and the amount lost from the system via various mechanisms, such as radiolysis, degassing and diffusion. Insufficient hydrogen (via the loss of hydrogen or incorrect hydrogen operating range), could result in a number of hazards, such as increased corrosion rates, and the formation of explosive mixtures of oxygen and hydrogen in the pressurisers due to radiolysis.

418. The RP has included two arguments relevant to the inclusion of hydrogen injection within the generic design of UK HPR1000 that support the claim "3.3.10 – the chemistry aspects of the plant design have been developed to reduce the nuclear safety risk SFAIRP". These are:

- hydrogen injection is designed to maintain a reducing environment and minimise PWSCC; and
- the hydrogen concentration is optimised to minimise the generation of radionuclides.

419. The RP proposes limits for hydrogen of between 10 and 50 $cm^3\ kg^{-1}$, and the operating range between [redacted] and 50 $cm^3\ kg^{-1}$ (Ref. 113). Although there is some variability, most international guidelines for PWRs recommend a normal operating range of between 25 and 50 $cm^3\ kg^{-1}$ (Ref. 75). The adoption of a different range therefore requires additional justification, and this forms part of my assessment in the following sections. As part of my assessment, I identified several factors which might impact on the choice of the hydrogen concentration operating range:

- The margin to the concentration required to suppress radiolysis.
- Whether the hydrogen dosing equipment was designed appropriately to control the hydrogen concentration to the selected level.
- Whether a lower concentration of hydrogen would affect the dosing pumps.
- Whether a lower concentration of hydrogen would impact on claims made regarding corrosion performance of the steels that are used in the primary circuit of the UK HPR1000.

420. I will assess these factors in the subsequent sections of this report. For this assessment, I have used various pieces of ONR guidance, including SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). In addition, I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment, which provide details of the use of hydrogen in various contexts (Ref. 6).

421. Elsewhere in this report, I assess whether the dosing equipment is capable of maintaining the specified range (sub-section 4.2.2) and consider the adequacy of the degasification systems (sub-section 4.4.3.2).

4.2.7.1 Effects of Radiolysis

422. One of the main functions of hydrogen addition to the primary circuit is to maintain a reducing environment, which has the effect of suppressing the radiolysis of water. Only a very low concentration of hydrogen is required to achieve this effect; the RP stated in the response to RQ-UKHPR1000-0697 that between 0.3 $cm^3 kg^{-1}$ and 5 $cm^3 kg^{-1}$ is required (Ref. 3). Such a low concentration would be difficult to control, and to suppress the effects of water radiolysis in the region of sub-cooled boiling, a lower limit of 10 $cm^3 kg^{-1}$ is proposed by the RP (Ref. 113). This limit should enable a sufficient margin to that required to suppress radiolysis, and I am content that it is reasonable.

423. To allow for an appropriate margin to this limit, the RP initially proposed that the normal operating range would have a lower limit of 17 $cm^3 kg^{-1}$. However, this is somewhat lower than that specified in international guidance and would lead to a slightly lower margin to that where hydrogen concentrations would be a concern. This was one factor which I raised in RQ-UKHPR1000-0847, which asked the RP to provide further evidence to justify the lower operating target of 17 $cm^3 kg^{-1}$ (Ref. 3).

424. As is discussed in subsequent sub-sections, following this review, the RP changed the normal operating window to between [redacted] and 50 $cm^3 kg^{-1}$, with an expected operating range of between 25 and 35 $cm^3 kg^{-1}$, and produced a Category 3 modification to introduce the change into the safety case (Ref. 107). In terms of providing sufficient margin to radiolysis, this change did not affect it materially. I consider how it affects corrosion and radiation fields in the subsequent sub-sections.

4.2.7.2 Effects on Corrosion

425. In terms of controlling corrosion, hydrogen is added to the primary circuit to aid in the prevention of SCC of stainless steels. As was mentioned in sub-section 4.2.7.1, by suppressing the formation of oxygen through radiolysis of water, the presence of hydrogen removes one of the key initiators of SCC (sensitised materials are susceptible to cracking in the presence of chloride and oxygen). The RP provides a selection of research and plant data to support the selected operating range (Ref. 113). By maintaining a hydrogen concentration of greater than 10 $cm^3 kg^{-1}$, the RP claims that the resultant electrochemical potential precludes SCC for stainless steels. The RP also presents data that demonstrates that beyond a concentration of about 35 $cm^3 kg^{-1}$, there is only a very small benefit of additional hydrogen; the crack growth rate does not decrease significantly. The RP proposes that this means that it is sensible to operate up to 35 $cm^3 kg^{-1}$, but no further, in terms of SCC mitigation. This is in agreement with a review of RGP that I commissioned during Step 3 of GDA; I am therefore content with the arguments for stainless steel corrosion protection (Ref. 109).

426. In addition to protecting stainless steels, the addition of hydrogen is beneficial in terms of nickel corrosion, although there is less of an effect when compared to stainless steels. The design of the UK HPR1000 uses Alloy 690TT, alongside the related weld materials Alloy 52 and 82. These materials have never been known to crack in service (Ref. 109). Therefore, there are no reliable data which can be used to support the development of hydrogen operating chemistry in terms of Alloy 690TT performance. The performance of Alloy 690TT is generally agreed to be superior in terms of corrosion performance than the materials that were used in earlier designs of plant, such as Alloy 600. To support the claims made in the case, the RP has therefore used evidence gained from operating with Alloy 600. Some evidence indicates that a much lower hydrogen concentration would be beneficial for Alloy 600, but as the RP argues, Alloy 690 and related weld materials have shown significantly improved properties, and

these materials do not exhibit a strong correlation of crack growth rates with hydrogen concentration.

427. In terms of crack initiation, the evidence presented generally supports the operating range selected by the RP, and shows that operating in the range between 25 and 50 cm^3 kg^{-1} does not significantly affect crack initiation. I am therefore content with the RP's arguments on the effect of hydrogen dosing on corrosion of the UK HPR1000 primary circuit materials.

4.2.7.3 Fuel Cladding Integrity

428. The RP presented evidence to support the claim that dosing hydrogen into the primary circuit is beneficial in terms of the M5 fuel cladding corrosion (Ref. 113). Corrosion of fuel cladding is a well-known phenomenon at the lower end of the hydrogen concentration operating range. However, at higher concentrations, hydriding of the cladding becomes a concern, and an upper limit is required to mitigate this risk. It is generally accepted that an upper limit of 50 cm^3 kg^{-1} of hydrogen is effective at precluding hydriding of the cladding material, and most international guidelines support this upper limit (Ref. 75).

429. In terms of a lower limit, the RP presented some experimental data in an attempt to show that lower hydrogen concentrations would be beneficial for the fuel clad, in terms of minimising the amount of deposits that would form on the fuel. Much of the evidence presented by the RP on this subject is not new, and has been assessed by ONR during previous GDAs (Ref. 114). This previous assessment concluded that the evidence was complex and a clear correlation between hydrogen concentration and fuel deposits was not apparent. It was also likely that there was no effect below 26 cm^3 kg^{-1}. In RQ-UKHPR1000-0847, I asked the RP if any more evidence was available that could strengthen this part of the case (Ref. 3). In response, the RP stated that this evidence suggested that operating in the lower end of the normal operating range resulted in less deposition, than at higher concentrations (towards 50 cm^3 kg^{-1}). As I have previously stated, the available evidence regarding hydrogen concentration and the precise solubility of various species at PWR operating conditions is not comprehensive, and the arguments presented by the RP I also judge to be fairly weak. In addition, during Step 3 of GDA, as part of a review of appropriate RGP, my TSC did not find any evidence to support operating at the lower end of the range (below [redacted] cm^3 kg^{-1}, (Ref. 109)). Whilst I do not judge that the evidence presented supports moving to a very low hydrogen concentration, the evidence does tend to support operating nearer to the lower end of the selected normal operating range, in terms of fuel cladding.

4.2.7.4 Effects on Radiation

430. Higher hydrogen concentrations tend to increase the solubility of nickel ferrite ($NiFe_2O_4$), but decrease the solubility of nickel metal. It is still not clear which will be the dominating factor for the UK HPR1000, but the very limited data in the literature would suggest that lower hydrogen concentrations (around 25 to 40 cm^3 kg^{-1}) result in decreased ^{58}Co (through activation of dissolved nickel), which would be beneficial (Ref. 109, Ref. 75).

4.2.7.5 Lower Hydrogen Concentration Operating Limit

431. To support the lower operating hydrogen concentration limit, the RP provided OPEX from a number of plants, from a range of sources in the publicly available literature (Ref. 113). In general, the evidence provided was quite limited, the data presented tended to be quite scattered, and I judged that it did not provide a clear demonstration to support the argument about the lower concentration limit. The RP also presented OPEX from several plants that have altered the hydrogen concentration during at-power operations. Again, the RP claims a benefit in terms of activity, but the data is

very limited, and the effect is not clear. Further evidence is presented from the Ikata reactor (a Japanese PWR), and the Chinese Tianwan VVER, and although a slight correlation can be discerned, no statistical analysis is presented, which would help to strengthen the argument.

432. Finally, the RP presented some evidence from CGN's fleet of reactors. The RP stated that the data was taken from a plant with the same materials of construction as the UK HPR1000; stainless steel, zirconium alloy (M5), nickel-based alloy (Alloy 690TT), and the corresponding weld materials (Alloys 52 and 152). Whilst controlling to a constant pH, the RP states that operating at a lower hydrogen concentration (in the range of 25 – 30 cm^3 kg^{-1}) saw lower ^{58}Co levels than when operating at higher hydrogen concentrations (30 – 40 cm^3 kg^{-1}), but I do not judge this evidence to be strong. I also note that the concentrations of hydrogen in this study are significantly higher than the operating range specified by the RP.

433. Following further queries that I made about this evidence, the RP decided to review the target operating range for hydrogen. The RP provided further evidence, including new information from EDF that a higher hydrogen concentration was desirable.

434. The RP presented some additional data about the solubility of nickel, nickel ferrite and nickel oxide (NiO), which depend on hydrogen concentration. A change in the hydrogen concentration will affect the solubility of these species, and a balance between competing factors must be found, with the aim being the prevention of deposition of metallic nickel in the core, a source of fuel deposits. Although the data is limited, the RP suggests that by operating towards the lower end of the proposed range ([redacted] –50 cm^3 kg^{-1}), nickel solubility is increased and NiO solubility is decreased, which would be beneficial in terms of avoiding nickel metal deposition on the fuel.

435. A new summary of plant OPEX was provided to support the revised operating range:

- CPR1000 plants in China (operating window of [redacted] cm^3 kg^{-1}, expected value of [redacted] cm^3 kg^{-1}.
- [redacted]
- In addition, this range is also in line with the expected operating range at [redacted].

436. Only a summary of the operating data from CGN plants and EDF is presented. Whilst this is to be expected of the EDF data, the RP was not able to share their plant data during GDA. This is related to Assessment Finding AF-UKHPR1000-0011.

437. The RP stated that it is of limited benefit to corrosion control to operate above 35 cm^3 kg^{-1}, in terms of corrosion performance of both stainless steels and nickel alloys, which I have reviewed in the previous sections. The RP also acknowledged that operating at lower hydrogen concentrations reduces the margin to where radiolysis suppression fails.

438. I judge that the new revised expected operating range of between 25 and 50 cm^3 kg^{-1} is better supported by the evidence provided by the RP, and it is consistent with RGP (Ref. 75, Ref. 109). I am therefore content that it is appropriate for the UK HPR1000 and represents the option that reduces risks SFAIRP. However, should further developments in this field occur, the licensee should maintain focus on this developing area.

4.2.7.6 Upper Hydrogen Concentration Limit

439. The RP presented a series of arguments to support the proposed 50 cm^3 kg^{-1} upper limit for hydrogen concentration:

- There is very limited available OPEX to support operating above 50 cm^3 kg^{-1};
- EPRI guidelines support an upper limit of 50 cm^3 kg^{-1}.
- At high hydrogen concentrations, Zircaloy-based components, such as the fuel clad, are capable of picking up hydrogen and forming a brittle hydride. OPEX from operating plants suggests that operating up to 50 cm^3 kg^{-1} does not present a risk of developing fuel cladding embrittlement.
- The design limit of 50 cm^3 kg^{-1} does not present a risk of hydrogen explosion, due to the mitigations provided in the design of the UK HPR1000.

440. I consider the risks of hydrogen explosions in sub-section 4.4.3.2. I am content that the proposed upper limit of 50 cm^3 kg^{-1} is consistent with RGP and the available OPEX (Ref. 75, Ref. 109).

4.2.7.7 Strengths

441. By the end of Step 4 of GDA, the RP had identified an operating range for hydrogen that tallied with RGP, and the RP has also set out a clear, coherent justification for the limits that have been specified (Ref. 109).

442. The RP has set out a clear set of arguments that explain the basis of the limits and conditions necessary in the interests of safety that have been identified, despite a limited evidence base.

443. The choice of hydrogen concentration operating range is a balance between competing factors. The RP has recognised this and adequately demonstrated how the chosen operating range means the risks are reduced SFAIRP.

4.2.7.8 Outcomes

444. Based upon the assessment of hydrogen addition in UK HPR1000 described in sub-section 4.2.7 above, I have not identified any Assessment Findings. I did identify an area that is relevant to the Assessment Finding concerning the inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011) raised earlier in this report, however.

445. I also did not identify any minor shortfalls.

4.2.7.9 Conclusions

446. The RP has demonstrated that the chosen hydrogen concentration in the primary circuit minimises risks SFAIRP, minimising corrosion and the generation of radioactivity, and minimising the risks of any potential detriments.

447. Based on the outcome of my assessment of hydrogen addition in UK HPR1000, I have concluded that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

448. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the generic safety case.

4.2.8 Primary Circuit Sampling

449. Primary circuit chemistry is important to a number of safety-related aspects of the generic UK HPR1000 design and, as such, it is essential that adequate control and

maintenance of the chemistry parameters is achieved. A significant step towards controlling the chemistry is to ensure that representative sampling and analysis is carried out at an appropriate frequency.

450. ONR's expectations for chemistry monitoring, sampling and analysis are set out in SAP ECH.4 and NS-TAST-GD-089, and note that adequate provisions should be in place to ensure that the required level of chemistry control for safety will be effectively implemented, including obtaining and maintaining the required quantity and quality of data to support decision making, throughout all phases of the operational life of a nuclear facility (Ref. 2, Ref. 6).

451. The RP makes a number of sub-claims and arguments relating to primary circuit sampling and monitoring in PCSR Chapter 21, including that the REN [NSS] is designed to achieve representative sampling for the chemistry and radiochemistry parameters (Ref. 5).

452. My assessment of the UK HPR1000 sampling systems has been informed by a TSC contract (Ref. 34).

4.2.8.1 Nuclear Sampling System

453. The REN [NSS] is described in the relevant SDM and DSR as a centralised system used to sample and monitor both liquids and gases for various chemical and radiochemical parameters (Ref. 115, Ref. 116). The REN [NSS] is comprised of three sub-systems: the primary sampling system, secondary sampling system and a post-accident sampling system. Sampling of the SGs and the secondary circuit is assessed in sub-section 4.3.4.3 of this AR.

454. The REN [NSS] design has the capability for both continuous and periodic sampling. Online analysers are used to continuously monitor boron, hydrogen and oxygen in the RCP [RCS], with the sample temperature, pressure and flow being adjusted to the working conditions of the online analysers. Periodic sampling relies on grab samples, with radioactive liquid and gaseous samples being sent to a centralised glove box to protect operators. All sampling components and lines of the REN [NSS] are made from stainless steel to reduce corrosion and contamination of samples.

Continuous Sampling

455. Online analysers are used to continuously monitor the concentrations of boron, hydrogen and oxygen in the RCP [RCS]. A single online analyser is used for each parameter, with the system designed so that primary coolant from several locations of the primary circuit can be directed to the online analysers. The following locations can be sampled:

- The hot leg of RCP [RCS] loop 2
- The pressurizer (liquid phase)
- The hot leg of RCP [RCS] loop 3
- Downstream of the RHRS heat exchangers
- Downstream of RCV [CVCS] charging pumps
- Upstream of RCV [CVCS] coolant purification

456. In early reviews of the REN [NSS] DSR, it was not clear which sampling lines would be used in different operating modes to provide the most appropriate samples (Ref. 116). I raised two RQs on this matter (RQ-UKHPR1000-1124 and RQ-UKHPR1000-1375), and in response the RP noted that during Reactor in Power and Normal Shutdown with Steam Generators (NS/SG) modes, primary samples are obtained from the RCP [RCS] (Ref. 3). When the primary system is depressurised with RCP [RCS] pumps shutdown, samples will be obtained from downstream of the RHRS heat exchangers, when the

RIS [SIS] is in RHR mode and connected to the RCP [RCS]. Normally the online analysers are intended to be connected to the sampling line of the RCP [RCS] loop 2 or 3; if there was a failure in the sampling of loop 2, the analysers would be switched to RCP [RCS] loop 3. The RP's response to the RQs provided some clarity, with loop 2 and 3 being established as the RP's preferred choice of sample location for the measurement of hydrogen, oxygen and boron concentration. In the DSR, it appears that dissolved oxygen and hydrogen are measured from the hot leg of RCP [RCS] loop 2, the pressurizer (liquid phase), the hot leg of RCP [RCS] loop 3, downstream of RCV [CVCS] charging pumps and upstream of RCV [CVCS] coolant purification (Ref. 116). It is unclear how and why the system will switch between these sample locations to provide the most relevant data, and although it appears the system will perform adequately, further clarification on the intended operation of the system would strengthen the safety case. I consider this to be a minor shortfall.

457. Unlike some PWRs, the generic UK HPR1000 design does not include safety classified boron meters to directly detect a change in boron concentration; the protection is instead provided via detection of changes in core parameters. I raised RQ-UKHPR1000-1375 due to a lack of detail in the REN [NSS] safety case submissions on the capability of the system to measure a boron dilution event (Ref. 3). The RP claims that if the boron meter detects deviations of 50 ppm boron, it will trigger an alarm, with the operator then required to make a decision on whether a boron dilution event is taking place; it was unclear from the submissions whether the boron sampling capability would be sufficient to detect the 50 ppm deviation quickly enough. The RP's response to the RQ provided information on the online analyser response time in relation to boron concentration during different leak scenarios from the reference HPR1000 design, noting that in the event of the online analyser being unavailable, grab sampling would be performed once per day in normal operation with the time required for manual sampling and analysis being less than half an hour. This frequency of sampling has the potential to miss a boron dilution event, however, the RP claims that there is no time requirement for boron monitoring in the incident of a boron dilution event and that monitoring is used only as a preventative measure. This is due to boron monitoring not being in the safe path and therefore no diversity requirement for a more rapid response has been considered (the primary means of detecting a boron dilution fault, as claimed in the fault schedule, is the core detection system). Further justification of this position and consideration of whether a more robust means of boron monitoring is appropriate would strengthen the safety case; I consider this to be a minor shortfall. Fault Studies colleagues have assessed boron dilution faults in more detail and raised a related Assessment Finding, for the licensee to consider whether it is reasonably practicable to provide automatic isolation of the boron dilution source via the REN [NSS] (Ref. 117). Other aspects of boron chemistry control are assessed in sub-section 4.2.2.4.

458. It is common in PWRs to monitor both pH and conductivity of the primary coolant via online monitors. However, the RP's primary water quality specification instead calls for monitoring of lithium, chloride, fluoride and sulphate for impurity control (Ref. 30). Radioactive coolant needs to be sampled via a grab sample once per day to monitor lithium concentration. Whilst I consider that the system will perform adequately in its current configuration, further justification would strengthen the RP's case, as the addition of an online lithium meter may reduce the operator's exposure to radiation. I consider this to be a minor shortfall.

459. Regarding representative sampling, the system is designed so that continuous flow can be maintained in the sample lines in order to obtain a homogenous sample. To ensure homogeneity, it is important to maintain a continuous turbulent sample flow by achieving a Reynolds number of greater than 4000. The RP provides Reynolds numbers for 5 pipe zones which are all equal to or greater than 4000. The samples for online analysers are taken by branch sampling lines; the RP notes that the branch

sampling line upstream of the boron meter is approximately 6 m in length, and when switching from different sampling lines the branch will be flushed to ensure a representative sample. There is evidence that American Society for Testing and Materials (ASTM) and EPRI guidelines have been considered with regards to the location of sampling points, sampling nozzles, flow rates, sample conditioning, material selection and sampling line length. I judge that appropriate measures have been considered by the RP to facilitate representative sampling.

460. During Cold Functional Testing (CFT) and before HFT the RP claims that there is no requirement for continuous surveillance via the REN [NSS]. The operation of the REN [NSS] is not essential and if not available, samples will be taken from local or temporary sampling points, such as drain pipes. The REN [NSS] will be available during HFT, and its operation will be the same as in normal operational conditions. Further justification of the sampling and monitoring requirements during pre-operational testing will be needed as CFT and HFT procedures are developed; I judge that this can be achieved as part of the normal business activities of the licensee. Other aspects of the commissioning of UK HPR1000 are assessed in sub-section 4.2.11.

Manual Sampling

461. The majority of periodic sampling is sent to a centralised glove box to protect operators from chemical and radiochemical hazards. Moreover, sampling lines for continuously transporting radioactive fluids are installed behind a biological shielding wall. Samples are divided into four categories with each category having an independent glovebox. The four categories are:

- Primary side radioactive liquid samples (online analyser sample line locations)
- Radioactive liquid samples (grab samples)
- Slightly radioactive samples from nuclear auxiliary systems
- CPs sampling

462. Manual samples are collected from a branch line in a similar fashion to the online analysers. Branch sampling lines are flushed, with waste transferred to the NI RPE [VDS]. After manual sampling has taken place, the glovebox is decontaminated with demineralised water. Gaseous samples, including dissolved hydrogen, dissolved oxygen, noble gases and other gaseous species, can be manually sampled through the gas-liquid phase separator which is also located in a glovebox. The gas is sampled and sent to the laboratory for composition analysis. Local manual sampling is also performed on the TEG [GWTS] and the RPE [VDS], which is claimed to be infrequent and therefore is not connected to the centralised system. A sampling tank will be used instead and is connected to the sampled system or component before being sent to the laboratory for analysis.

463. The CPR1000 design is often referred to in UK HPR1000 sampling documentation, since the sampling and monitoring systems in the two designs are very similar. Whilst the RP notes that there has been no negative feedback from the operating CPR1000 plants to indicate that the current sampling and monitoring design cannot reflect the plant state, very little plant data or other OPEX from CPR1000 or other PWRs is presented in the submissions. The generic safety case instead relies on a comparison of the sampling system design features with international standards, not all of which are specific to nuclear systems. A lack of OPEX in the safety case is particularly relevant in the case of the design for zinc sampling and monitoring, which is neither justified by the referenced standards nor by reference to the CPR1000 design (which does not dose zinc). Zinc readily incorporates into the sample line metal oxide film making it difficult to measure accurately and consistently. In UK HPR1000, zinc is sampled via grab sampling from the hot leg of RCP [RCS] loop 2, the hot leg of RCP [RCS] loop 3 and the RCV [CVCS].

464. The sampling locations chosen appear reasonable, and the RP appears to have taken suitable steps in order to facilitate representative sampling of zinc, such as reducing sample line lengths, lowering the sample temperature as soon as possible after sample take-off from the main system, and maintaining continuous sampling flow. However, as zinc dosing is a new feature to the generic UK HPR1000 design (assessed in sub-section 4.2.6), the presentation of OPEX from other similar plants to justify that the system can effectively sample and analyse for zinc would improve the safety case. Overall, I judge that the safety case would be improved if plant data or other OPEX (from CPR1000 or other PWR designs) was presented to strengthen the justification that the REN [NSS] can provide representative sampling for all chemistry parameters, but with a particular focus on zinc since achieving representative zinc sampling can be a particular challenge; I consider this to be an Assessment Finding. This is also linked to AF-UKHPR1000-0011 on OPEX to support the safety case.

AF-UKHPR1000-0159 – The licensee shall, as part of detailed design, justify the capability of the nuclear sampling system to perform representative sampling for zinc in the primary coolant.

465. For representative particulate sampling, isokinetic sampling probes are often used. The RP has recognised that this is common practice but has opted not to use this technology in CPs sampling, due to the low proportion of particulate in the primary coolant. Although my TSC did not consider that the CPs analysis will be greatly affected by the omission, I consider that insufficient evidence has been presented in the current safety case to justify the decision not to use an isokinetic sampling probe. Further justification by the licensee will improve the case in this area, and hence I consider this to be a minor shortfall.

4.2.8.2 Post-Accident Sampling System

466. As described in more detail in sub-section 4.5, in the event of an accident, radiolysis products (hydrogen and oxygen), steam, particulates and radioactive fission products (noble gases, radioactive aerosols and iodine) may be present in the reactor containment atmosphere. Sampling and analysis of the atmosphere can give an indication of the temperature history of the core during the accident and provide information on the physical state of the core. Early information concerning the extent of the damage and the conditions in the containment after an accident is important in informing subsequent accident management measures.

467. The RP claims that the sampling lines of the RCP [RCS] loops and the downstream RHRS heat exchangers and SG blowdown lines are available for use under post-accident conditions. Furthermore, there is a dedicated sub system, REN9, that is specific to post-accident sampling and collects liquid and gaseous samples from within the containment. Sampling points include the containment atmosphere and sump, the IRWST, and the Containment Heat Removal System (EHR [CHRS]). The REN9 sampling equipment is located in a separate building close to the reactor building. To protect operators, the room is spilt into two using lead shielding, with a sampling process module located behind the lead shielding and the operational panel in front of the shielding wall. Sample collection and processing are handled through the sampling process module. The system is set up to deal with both liquids and gases, with automatic dilutions of samples being performed to reduce the radioactivity by a factor of 500, 2000 or 20000 before being transported to a laboratory for analysis.

468. The justification of why post-accident sampling is required is also described, referring to IAEA standards; NS-G-1.10, NS-G-1.13, SSR-2/1, and SSG-13 (Ref. 11, Ref. 9). Although this is helpful, the documentation provided does not elaborate on how the system requirements from the standards and guidance have been implemented into the REN9 system design. Further description of these aspects would strengthen the

RP's case; I consider that this information can be provided by the licensee as part of normal business.

469. Overall, the design of the system appears to be adequate for post-accident sampling. Further information will be needed on the procedures for post-accident sampling, and to further justify the capability of the system, as the safety case develops, however I consider that this information can be developed by the licensee as part of normal business.

4.2.8.3 Strengths

470. Overall, I note that a number of beneficial features have been included in the design of the primary circuit sampling systems, including:

- Online analysers for several important primary circuit parameters to detect any sudden transitions in a timely manner.
- Development of the sampling system design and layout in line with RGP in order to facilitate representative sampling. For example, sampling lines are as short as practicable, flow rates and flushing times have been considered and stainless steel has been chosen for pipe materials.

4.2.8.4 Outcomes

471. Based upon the assessment of the primary circuit sampling systems in UK HPR1000, I have identified one Assessment Finding concerning the justification of the capability of the NSS to perform representative sampling for zinc in the primary coolant (AF-UKHPR1000-0159). I have also identified an area that is relevant to the Assessment Finding concerning inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011) raised earlier in this report.

472. I also identified a number of minor shortfalls, as described in sub-section 4.2.8 above.

4.2.8.5 Conclusion

473. Based on the outcome of my assessment of the primary circuit sampling systems in UK HPR1000, and in line with SAP ECH.4, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA (Ref. 2). For the majority of chemistry and radiochemistry parameters, the RP has provided an adequate justification to support the claim that representative sampling can be achieved.

474. I have identified one Assessment Finding as a result of this part of my assessment.

4.2.9 Holistic Approach to Normal Operating Chemistry

475. As I have described in the previous sections of this report, the primary circuit normal operating chemistry has many different requirements, and represents a balance of many competing factors. This balance of factors is required as the chemistry regime has the potential to influence phenomena such as waste arisings, ORE, the structural integrity of the SSCs of the primary circuit and supporting systems, and the integrity of the fuel. The design of the primary circuit operating chemistry builds upon many years of operation of PWRs. Many of those PWR units have adopted the familiar basis of a coordinated lithium/boron chemistry to achieve the target pH, based upon reactivity considerations. This has been combined with the close control of oxygen and the addition of hydrogen, to maintain a reducing environment within the circuit. Finally, deleterious species, such as chloride and sodium have been tightly controlled. A summary of the primary circuit chemistry is shown in Annex 3.

476. For the UK HPR1000, recent developments in primary circuit chemistry have also been incorporated into the design, such as the use of EBA, and the choice to inject zinc, in the form of DZA), in an effort to reduce corrosion and ORE. In the previous sections of this report, I have assessed each component of the primary circuit chemistry regime in detail, but it is also important to assess the overall regime holistically, to assure that an appropriate overall balance is struck. In undertaking this assessment, I have used the guidance set out in SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). However, more significantly, I have used the guidance supplied in SAP ECH.2, which requires the resolution of conflicting chemical effects so that an appropriately balanced safety case has been developed. SAPs EAD.1 and EAD.2 are also relevant for the minimisation of corrosion. I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6).

477. As I have already set out, the basis of the primary circuit chemistry regime is as follows:

- A target $pH_{300\,°C}$ of 7.2 as soon as possible following start-up. This is achieved with a lithium hydroxide concentration of up to 3.5 mg kg^{-1} at the BOC, and the use of EBA. This is assessed in sub-section 4.2.3.3.
- A target operating window for hydrogen of between [redacted] and 50 cm^3 kg^{-1}. This aspect is assessed in sub-section 4.2.7.
- Impurities, such as chloride, sulphate and fluoride are minimised to a target of less than 150 μg kg^{-1} (Ref. 32). This aspect is assessed in sub-section 4.2.3.3.
- Zinc is added to the circuit, with a target operating concentration of between [redacted] and [redacted] μg kg^{-1}. This aspect is assessed in sub-section 4.2.6.

478. Although the GDA process does not require assessment of precise limits (these can be defined during the site-specific stages), assessment during GDA should include a judgement of the operating range where these limits will be defined. However, the RP has also proactively shared a significant amount of detail about the proposed limits and operating ranges, and although these may change during later phases of the safety cases development, they provide a useful basis for a judgement on the overall regime. The UK HPR1000 primary circuit controls are therefore as shown in Table 8 (Ref. 30).

Table 8: Primary Circuit Water Chemistry and Radiochemistry Limits during Power Operation

Control Parameter	Unit	Operating Window	Limit
Total B	mg kg^{-1}	As specified by fuel limits	As specified by fuel limits
^{10}B Enrichment	at%	35-39	-
Lithium Hydroxide	mg kg^{-1}	As specified by fuel limits	<3.5
Hydrogen	cm^{3} kg^{-1}	[redacted]50	10-50
Chloride	mg kg^{-1}	0.15	1.5
Fluoride	mg kg^{-1}	0.15	1.5
Sulphate	mg kg^{-1}	0.15	1.5
Zinc	µg kg^{-1}	[redacted]	[redacted]
Nickel	µg kg^{-1}	Site-specific stage	6
Silica	µg kg^{-1}	0.6	1
^{131}Ieq	GBq t^{-1}	Site-specific stage	16
Σ_{gas}	GBq t^{-1}	Site-specific stage	[redacted]
Δ^{133}Xe	MBq t^{-1}	[redacted]	-
^{133}Xe/^{135}Xe	-	[redacted]	-
Δ^{134}I	MBq t^{-1}	[redacted]	[redacted]

479. This list of limits and conditions necessary in the interests of safety is perhaps more detailed than expected at this stage of the development of the UK HPR1000, but it is welcome, and demonstrates the depth of understanding that the RP has developed in defining the primary circuit chemistry regime. Although some of these values may change somewhat during site-specific stages, this detail provides confidence that the regime has been adequately considered, and the licensee will be able to use this information to adequately define all relevant limits and conditions necessary in the interests of safety.

480. Through the assessment work I have undertaken in the preceding sections, and through interactions with the RP, I have gained sufficient confidence that in developing these limits and conditions necessary in the interests of safety, the RP has given adequate consideration to the interaction of the many factors that are necessary when defining the chemistry regime. This is an iteration of previous chemistry regimes, but the RP has demonstrated the technical understanding how these limits will affect aspects such as corrosion, radioactivity and fuel integrity, and balanced these competing factors appropriately.

4.2.9.1 Strengths

481. The RP has set out a coherent, logical approach to the overall primary circuit chemistry regime, which balances competing effects in an effort to minimise corrosion, operator dose and the risk of fuel deposits occurring. The RP has also adequately developed the generic safety case to demonstrate that this balance reduces risks SFAIRP.

4.2.9.2 Outcomes

482. Based upon the assessment of the holistic approach to normal operating chemistry in the UK HPR1000, described in sub-section 4.2.9 above, I have not identified any Assessment Findings.

4.2.9.3 Conclusion

483. Based on the outcome of my assessment of the holistic approach to normal operating chemistry in the UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

484. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the safety case.

4.2.10 Start-up and Shutdown Chemistry

485. At the end of each fuel cycle all PWRs shutdown for refuelling and maintenance and, when this is completed, return to normal operating conditions during a start-up. Over the decades, much effort has been made to optimise the chemistry control during the various modes of operation that the terms shutdown and start-up cover. Badly conceived practices have in the past led to increased operator doses, and greatly lengthened outage periods. However, despite significant improvement, no universal approach is applied to PWRs, mainly due to plant differences, or differences of approach to the technical basis.

486. A general description of the steps that are made during a shutdown can be summarised as follows, with many steps acting concurrently:

- Boration (to cold shutdown concentration)
- Cooling and depressurisation of the RCP [RCS]
- Dehydrogenation and oxygenation
- Purification
- Primary circuit opening

487. During start-up, these steps occur in reverse order, with the exception of purification and the addition of lithiation. To be able to perform these steps, several separate systems are required to operate throughout the entire sequence. The assessment of start-up and shutdown in this report has concentrated on the chemistry changes and the controls on these, rather than a system by system assessment. However, many of the most important systems are assessed elsewhere in this report.

488. A number of significant chemistry changes take place during the transients associated with both start-up and shutdown. The primary circuit is taken from hot reducing alkaline conditions (during normal operation), to cold oxidising acidic conditions (during shutdown), and back again. These changes cause several effects; during shutdown

the most important of these is the intentional increase in the concentrations of both soluble and particulate radionuclides (from fuel deposits and soluble CPs) in the coolant. Hydrogen peroxide is added to highly borated water to allow for the dissolution of deposits within the primary circuit, which can then be removed prior to start-up. This is performed to improve the performance of the plant in subsequent cycles, and to reduce operator doses. This change 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 and maintaining adequate chemistry control.

489. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment, which provide some guidance on factors to consider for start-up and shutdown (Ref. 6).

4.2.10.1 Overall Approach to Start-up and Shutdown

490. Two of the main sub-claims in the safety case are relevant for all modes of operation (Ref. 5):

- 3.3.10. SC21.1: The primary chemistry and process are optimised in all operating modes to maintain the integrity of the safety barriers in the primary circuit.
- 3.3.10. SC21.2: The chemistry regime supports the control of reactivity in all operating modes.

491. There are also two sub-arguments that are relevant to start-up and shutdown chemistry, which state that the hydrogen concentration is controlled during shutdown to minimise the risk of a hydrogen explosion, and that the pH and redox environment of the primary coolant are controlled to accelerate the decomposition and dissolution of deposits and minimise the duration of shutdown.

492. I conducted a high-level review of the RP’s submission against the expectations made in common chemistry guidance (Ref. 75, Ref. 118). I consider that the proposals set out by the RP generally meet the expectations set out in the guidelines. However, I noted the following gaps:

- The RP did not specify a head-lift criteria for ^{3}H. This is contrary to much international OPEX, and indeed within the UK. The general approach to ^{3}H, including relevant limits and conditions necessary in the interests of safety, is discussed in sub-section 4.2.3.3.
- Prior to shutdown, no reduction in hydrogen concentration was described.
- During shutdown, the ‘Topic Report on Start-up and Shutdown’ contained very little information about the control of the oxidation process, and how hydrogen peroxide is added to the circuit, to ensure sufficient radionuclides are removed via the dissolution of corrosion products.
- Following the oxidation step, mechanical degassing of the circuit is followed by chemical degassing, using hydrazine. However, few details were described in the safety case about how this operation is controlled, and how the plant is protected from excessive concentrations of hydrazine decomposition species, such as ammonia interacting with the resins in the RCV [CVCS].

4.2.10.2 Control of Water Chemistry

493. The RP describes the start-up and shutdown requirements of the primary circuit in terms of reactivity control, pH control, redox environmental control and impurity control (Ref. 29). The operating modes are as described in Table 9:

Table 9: UK HPR1000 Operating Modes During Start-up and Shutdown

Normal Operating Modes	Standard Operating Conditions	RCP [RCS] State	RCP [RCS] average Temperature (°C)	RCP [RCS] pressure (bara)
Reactor in power	Reactor in power	Closed	[redacted]	[redacted]
	Hot standby	Closed	[redacted]	[redacted]
Normal Shutdown with Steam Generators (NS/SG)	Hot shutdown	Closed	[redacted]	[redacted]
	Intermediate shutdown with NS/SG connection Conditions (P ≥130 bara)	Closed	[redacted]	[redacted]
	Intermediate shutdown with NS/SG connection Conditions (P< 130 bara)	Closed	[redacted]	
	Intermediate shutdown with RIS-RHR connection conditions	Closed	[redacted]	[redacted]
Normal shutdown with RIS-RHR (NS/RIS-RHR)	Intermediate shutdown with RIS-RHR	Closed	[redacted]	[redacted]
		Closed	[redacted]	[redacted]
		Closed	[redacted]	[redacted]
	Normal cold shutdown (RCP [RCS] pressurisable)	Non-closed and pressurisable	[redacted]	[redacted]
Maintenance cold shutdown (MCS)	Normal cold shutdown for maintenance (RCP [RCS] not pressurisable)	Non-closed and not pressurisable, Reactor cavity non fillable	[redacted]	[redacted]
Refuelling cold shutdown	Normal cold shutdown for refuelling	Non-closed and not pressurisable, Reactor cavity fillable	[redacted]	[redacted]

Reactor completely discharged (RCD)	Core totally unloaded	-	[redacted]	[redacted]

494. Prior to the addition of hydrogen peroxide, the hydrogen concentration is required to be less than 3 $cm^3 kg^{-1}$, a reduction from the normal operating range of at least [redacted] $cm^3 kg^{-1}$. This is done to minimise the amount of hydrogen when the circuit is opened up, and to minimise the risk of Low Temperature Crack Propagation (LTCP) from occurring, which I will discuss later in this section. However, the safety case did not describe how this concentration would be changed, over what period, and when prior to shutdown it would occur. The RP clarified that the hydrogen concentration should be reduced prior to the first stage of shutdown to about [redacted] $cm^3 kg^{-1}$, which was a sufficient concentration to achieve an acid reducing environment to promote decomposition of nickel ferrite in the NS/SG mode (Ref. 113). The RP also stated that the TEP [CSTS] degasification unit is put into operation about 24 hours before starting the shutdown process, to reduce the hydrogen concentration down to the lower operating window concentration ([redacted] $cm^3 kg^{-1}$), which is monitored by the REN [NSS]. Following this, the hydrogen concentration is reduced to below 3 $cm^3 kg^{-1}$ using the degasser, prior to the forced oxygenation phase. I am satisfied that this response, which aligns with the guidance I have reviewed (Ref. 118).

495. Following the issue of RQ-UKHPR1000-0460, the RP updated the ‘Topic Report on Start-up and Shutdown’ to include information about the control of the dissolution and oxidation process to remove CPs during shutdown, to ensure sufficient radionuclides are removed (Ref. 3).

496. During the initial acid reducing environment that allows for the dissolution of CPs during shutdown, nickel ferrite and magnetite (Fe_3O_4) are dissolved. Nickel ferrite reduces under dissolved hydrogen to produce nickel metal. ^{58}Co is associated with nickel, and derives from nickel sites in the fuel deposits. The process to dissolve magnetite is a function of temperature, hydrogen and pH.

497. The RP states that by maintaining the expected dissolved hydrogen concentration for reductive decomposition of nickel ferrite near the lower operating window limit of [redacted] $cm^3 kg^{-1}$ during the NS/SG mode, the divalent nickel content of the ferrite can be reduced to nickel metal, and some of the trivalent iron can be reduced to divalent iron. Therefore, ^{58}Co contained in the fuel deposits (activated from nickel deposits) is released during shutdown and captured by the RCV [CVCS].

498. The RP also stated that the hydrogen peroxide is dosed via the RCV [CVCS] during shutdown when the coolant temperature is decreased to 80 °C. The dissolution of nickel proceeds rapidly in the acid-oxidizing condition of cold shutdown. Nickel metal that was produced by the reductive decomposition of nickel ferrite will be dissolved, along with any nickel metal that may have been present in the original fuel deposits. Prior to start-up, the nickel concentration has been specified to be below [redacted] $mg kg^{-1}$, to prevent reaction with hydrogen producing hydride species.

499. According to OPEX from the RP, the dosing period is expected to be about ten minutes, which will quickly oxygenate the coolant. Following this period, the clean-up systems will be put into operation when the oxygen content is about 1 $mg kg^{-1}$. The RP also said that iron, chromium, nickel and ^{58}Co will be monitored, and that the RCP [RCS] will not be opened until the ^{58}Co content is reduced to within the radiochemical specification, which has a limit of 1500 $MBq t^{-1}$; a limit set to minimise ORE (Ref. 31). These responses help to explain the processes and controls that govern this important part of the shutdown operation, which can be built upon by the licensee.

500. However, I did not consider that an appropriate justification that risks had been reduced SFAIRP had been made by the RP; the responses provided during most of Step 4 of GDA were essentially a list of operations that were to be followed, and some relevant limits. One of the main objectives of this part of the safety case is to justify that the chemistry regime chosen for start-up and shutdown reduces the level of radioactivity SFAIRP, and the safety case did not adequately describe how the chosen chemistry and operations would meet that goal, in terms of activity removal and a justification about activity transport, deposition and dissolution. This is part of Assessment Finding AF-UKHPR1000-0115.

501. LTCP is a term used to describe low temperature SCC that occurs at a pre-existing sharp crack induced by hydrogen embrittlement. It is a concern for nickel-based alloy materials, and therefore potentially could be a mechanism of concern for the UK HPR1000. The following conditions are required to occur simultaneously for LTCP to occur:

- relatively high concentrations of hydrogen;
- low temperatures (<150 °C);
- the presence of a sharp crack tip; and
- sustained high loads. The loads must be able to create stress in metal which approached the yield stress.

502. Although LTCP has not been experienced in operating plants, it has been studied extensively in laboratory testing. In response to RQ-UKHPR1000-0486 which I raised about the possibility of this phenomenon occurring in the UK HPR1000, the RP responded by stating that no occurrences of LTCP had been recorded in service at PWR-relevant conditions (Ref. 3). Additionally, the RP presented some data from tests that used Alloy 690, Alloy 52 and 82 weld metals, over a range of temperatures and hydrogen concentrations, and the results indicated that these materials did not show significant susceptibility to LTCP in water with dissolved hydrogen over 15 cm^3 kg^{-1} in 50 °C water. The RP confirmed that following an investigation of the operating conditions of the UK HPR1000, there was no possibility of the plant operating in 50 °C water with hydrogen concentrations greater than 15 cm^3 kg^{-1} (prior to shutdown the hydrogen concentration is lowered to around [redacted] cm^3 kg^{-1}). I also note that in the review of start-up and shutdown by my TSC, LTCP was not highlighted as a significant concern (Ref. 119). I am therefore content that LTCP will not be a concern for the UK HPR1000.

4.2.10.3 Degassing

503. Following the oxidation step in the shutdown process, mechanical degassing of the circuit is followed by chemical degassing, using hydrazine. However, few details were described in the safety case about how this operation is controlled, and how the plant is protected from excessive concentrations of hydrazine decomposition species, such as ammonia, which can interact deleteriously with the resins in the RCV [CVCS]. In the response to RQ-UKHPR1000-0709, the RP provided further information, and stated that the purification unit of RCV [CVCS] should be bypassed when hydrazine is injected to the primary circuit (Ref. 3). After the oxygen concentration in the primary circuit meets the chemistry control requirement, the excess hydrazine and ammonia can be removed through the exhaust pipeline of the pressurizer. The hydrazine and ammonia concentrations are monitored by the REN [NSS]. When the hydrazine and ammonia concentration is less than [redacted] µg kg^{-1}, the purification unit of RCV [CVCS] can be put into operation again. In the response to RQ-UKHPR1000-1450, the RP also confirmed that these are considered as formal limits in the safety case, and are based upon OPEX from Chinese plants (Ref. 3). I was satisfied with this response.

504. I note that in reply to RQ-UKHPR1000-0709, the RP states that in start-up and shutdown, the REA [RBWMS], is isolated, and the IRWST is used to provide boronated

water make-up to the primary circuit. This is a measure to prevent accidental dilution faults, as the REA [RBWMS] contains the demineralised water make-up and boric acid make-up pipelines.

505. Unlike most PWRs, the UK HPR1000 uses the IRWST to fill the RHRS prior to connection to the primary circuit. As this water is not maintained under a reducing environment, this approach leads to the introduction of a quantity of oxygenated boric acid into the circuit, although it also prevents a possible source of lithium contamination, which could cause a pH increase during shutdown. In response to RQ-UKHPR1000-0709, the RP confirmed that the approach taken in the UK HPR1000 to remove the dissolved oxygen that is introduced into the primary circuit via this means is to inject hydrazine into the primary circuit about one hour prior to connection of the RHR, via the RCV [CVCS], (Ref. 3). This process is carried out at a temperature of between 135 and 140 °C, which is consistent with the start-up procedure, and should present minimum risk to the primary circuit materials. I was content with this response, and I also note that this arrangement is similar to the design of the EPR™.

4.2.10.4 Control of Radioactivity

506. The RP describes the general methods used to remove radioactive species, both dissolved, particulate and gaseous, from the primary circuit and has identified limits for some of them. Dissolved and particulate species are removed by the RCV [CVCS] demineralisers and filters, and when the letdown flow from the primary coolant is sent to the TEP [CSTS] for recycling, further purification is available in the TEP [CSTS] purification station. Gaseous species are removed by the TEP [CSTS] degasifier column, and the VCT gas space is continually purged with nitrogen from the TEG [GWTS] system. These systems are assessed in more detail in sub-sections 4.2.2 and 4.4.3.

507. The RP has decided to place limits on the parameters, as displayed in Table 10 which also gives details of the limits that the RP has determined. The description of limits associated with ^{3}H is assessed in more detail in sub-section 4.2.3.3.

Table 10: Operating Limits for Active Species During Start-up and Shutdown

Parameter	Operating Window (MBq t^{-1})	Limit (MBq t^{-1})	Operating Mode
^{58}Co	[redacted]	[redacted]	Reactor Pool Flooding Completed
Total gamma	-	[redacted]	
^{133}Xe	-	[redacted]	Before Reactor Pressure Vessel Opening
^{131}I	-	[redacted]	
^{58}Co	[redacted]	[redacted]	
Total gamma	[redacted]	[redacted]	

508. These limits generally agree with those described in international guidance, and I am content that they are appropriate for GDA (Ref. 75).

509. Although the RP does provide shutdown and transient activity concentrations for each isotope, the RP has not identified the total peak in radioactivity that will occur during shutdown (Ref. 66). Neither has it described how the limits described previously are supported by this analysis. Although the RP has provided a justification that risks

associated with the radioactivity generated during start-up and shutdown has been reduced SFAIRP, I consider this to be a minor shortfall (Ref. 44).

4.2.10.5 Strengths

510. The RP has outlined a strategy for start-up and shutdown that generally adheres to RGP, and although detailed procedures will be produced during the site-specific stages, I am confident that the licensee will be able to successfully build upon the strategy produced by the RP.

511. The description of proposed limits for several radiochemical species is a welcome inclusion in the generic safety case.

4.2.10.6 Outcomes

512. Based upon the assessment of the start-up and shutdown philosophy of the primary circuit in UK HPR1000 described in sub-section 4.2.10 above, I did not identify any Assessment Findings. I did, however, identify one aspect that relates to an Assessment Finding raised earlier in this report, concerning the demonstration of how operating practices have been optimised to reduce the risks associated with the generation, transport and accumulation of radioactivity SFAIRP (AF-UKHPR1000-0115).

513. I identified one minor shortfall.

4.2.10.7 Conclusion

514. The RP has adequately justified the approach taken to control the primary circuit chemistry during start-ups and shutdown. Despite the need for work to demonstrate that these processes have optimised, the RP has demonstrated that the overall approach is reasonable, and in-line with RGP.

515. Based on the outcome of my assessment of the start-up and shutdown philosophy of the primary circuit in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

516. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the safety case.

4.2.11 Commissioning Chemistry

517. Commissioning of a reactor (as other complex industrial facilities) is a lengthy process that involves testing and confirmation of the operability of each of the reactor systems and components. From a Chemistry perspective, chemistry plays an important role in this process and involves activities such as surface cleaning and conditioning.

518. HFT is one aspect of the commissioning process, occurring in the later stages. HFT is a unique period of reactor operations, as it represents the first occasion(s) when the reactor is operated under full temperature and pressure conditions, albeit without the fuel. The chemistry adopted during this period is generally accepted to be important in determining the subsequent behaviour of the reactor, especially the primary circuit, in the ensuing fuel cycles (for example, shutdown releases and susceptibility to degradation mechanisms). It also has a pronounced effect on the subsequent ORE

and waste performance of the reactor over many subsequent cycles, if not the lifetime, of the plant (Ref. 120).

519. The understanding of the processes and requirements for HFT has evolved over recent years (and will continue to do so). Significant international experience is expected to influence the final choices (especially from any plant which may commission before any UK HPR1000 plant is licensed). The precise conditions and timings for such periods are greatly influenced by commercial constraints and licensee choices. For GDA, it is therefore not reasonable to expect the RP to have fully developed commissioning methods and procedures. My expectations and scope of assessment during Step 4 of GDA concentrated on the principles and the basis for the approach suggested, the availability and capability of the supporting systems required, identification of areas where further development may be required and a demonstration that the generic UK HPR1000 design has not foreclosed options that may have significant benefits for longer term plant safety.

520. The chemistry provisions for the commissioning phase for the primary and secondary circuit is included in Sections 4.3.2 and 5.3.2 of PCSR Chapter 21 respectively as follows (Ref. 5):

- ■ Primary circuit:
    - The chemistry regime during HFT (optimised for passivation of the SG surfaces)
    - Water quality control
    - Consumable requirements (in line with RCC-M F6400)

- ■ Secondary circuit:
    - Condition before HFT (layup conditions)
    - Condition during HFT (as normal operation)

521. The RP makes a number of arguments and sub-arguments in Chapter 21 of the PCSR for the chemistry aspects of the commissioning phase, including that passivation during hot functional tests is undertaken to minimise general corrosion (Ref. 5).

522. The main submission that supports the above claims and arguments is the ‘Topic Report on Commissioning Chemistry’ (Ref. 121). This submission is supplemented by other chemistry reports, in particular ‘Topic Report on Zinc Injection in the Primary Circuit of UK HPR1000’ and ‘Impact of Zinc Injection on Corrosion Resistance of Primary Circuit Material’ (Ref. 106, Ref. 108).

523. The principal SAPs relevant to my assessment of commissioning chemistry are ECH.1, ECM.1, EMC.16, EAD.1 and EHT.5, together with NS-TAST-GD-088 and NS-TAST-GD-089 (Ref. 6). I also raised two RQs, RQ-UKHPR1000-1079 and RQ-UKHPR1000-1459 to gain further clarification and evidence to support the claim made in the PCSR (Ref. 3).

524. The RP’s approach to the commissioning stage is to separate it into two phases, which are further defined by stages and sub-stages as noted below. The definition is based on the different risks and hazards, their potential significance, and the availability of key plant systems in keeping with the guidance set out in the SAPs ECM.1, EMC.16 and IAEA commissioning guidance (Ref. 2, Ref. 10).

- ■ Pre-operational test phase (inactive)
    - Stage I: Preliminary tests
    - Stage II: Functional tests (cold and hot functional tests)

- ■ Initial start-up test phase (active)

- Stage III: Initial start-up tests

525. The RP presents a breakdown of the major tests typically performed and the purpose of each sub-stage proposed for commissioning of the UK HPR1000 which are directly related to chemistry control activities during commissioning (Ref. 121). It also provides information on the required availability and capability of relevant UK HPR1000 systems and the typical chemistry controls and measurements that need to be taken during the various commissioning activities.

526. The RP's requirements and measures for cleanliness during the commissioning Stage I are centred around the guidelines within RCC-M (Ref. 54). Stage II and III typically align with normal operating procedures for water quality and impurity control. It is neither possible, nor reasonable, to present all the details here, however, given the particular importance of HFT for conditioning the primary circuit, in particular the SG tubes, further details of this aspect of my assessment are included here.

527. The proposed HFT procedure is based upon a three step process, which is similar to the steps taken during a normal operating cycle and plant shutdown, namely:

- Alkaline reducing – lithium hydroxide, zinc and hydrogen addition. This step results in the formation of the protective oxide on surfaces.
- Acid reducing – delithiation, addition of boric acid.
- Acid oxidising – forced oxygenation with hydrogen peroxide. This step aims to remove labile outer oxide material before power operation in order to prevent subsequent activation.

528. The RP recommends a minimum of 300 hours (12.5 days) under alkaline reducing conditions, noting that any increase in duration will have additional benefits in producing a more stable oxide with higher levels of zinc incorporation. The chemistry changes required are accommodated by the normal RCP [RCS] chemical control systems, such as the RCV [CVCS], which is assessed elsewhere in this report (sub-section 4.2.2), so are not repeated here. The chemistry requirements are generally akin to those expected during normal operations, so my assessments undertaken in these areas remain valid.

529. The application of a forced acidification step with boric acid, followed by a controlled oxidation during cool down to remove labile outer oxide deposited material are recognised steps for RGP in the RP approach. Removal of the outer layer oxides is an important step before later commissioning stages, which require boric acid addition due to the introduction of fuel. I welcome these additions and believe that an adequate case has been made for their implementation based upon the current knowledge of HFT processes.

530. In general, the approach suggested by the RP is consistent with the latest worldwide plant HFT RGP. The main differences to the RP procedure tend to centre on the duration and precise concentrations of the procedure (for example, the lithium hydroxide and hydrogen concentrations and the time pre-oxidation is allowed to take), however the fundamental process of hot reducing alkaline pre-oxidation appears well supported and suggests substantial reductions in CPs and associated dose to workers are possible. As per the Step 4 assessment objectives, the precise definitions of these values were not considered during GDA. However, the RP has presented well-reasoned arguments for the expected values and I particularly welcome the addition of hydrogen (as opposed to a reliance on the residual and variable hydrogen generated by oxidation) and expect this to offer further benefits.

531. The main difference in the RP's approach from previous HFT procedures, and that of the reference plant (FCG3), is the addition of zinc. The RP has specified a primary circuit chemistry for UK HPR1000 which includes the addition of zinc for radiation field

and corrosion control. Primary circuit chemistry and zinc addition are discussed in sub-sections 4.2.2 and 4.2.6 respectively. In the context of its injection during HFT, whilst the addition of zinc is strongly supported by experimental data for reduced corrosion rates (as discussed in sub-section 4.2.6), it is not supported by significant plant experience, with only a few reactors undergoing HFT with zinc to date (Angra 2 in 2000, Tomari 3 in 2008, Watts Bar 2 and Haiyang in 2016, Sanmen in 2018). The RP quote the example of the Japanese Tomari 3 plant to support injection of zinc during HFT; initial indications considering the first two cycles suggest significant improvements in plant dose and corrosion rates have been achieved when compared to plants operating without zinc (Ref. 122).

532. The application of zinc during HFT appears a reasonable proposal; the action of zinc primarily affects the inner oxide layer, which is precisely where HFT is aiming to have the greatest effect. It is not apparent how much zinc would be needed to fully saturate the RCP [RCS] surfaces however, and so careful consideration would be required as this may require a greatly increased HFT duration or much higher concentrations than during normal operation to offer significant benefit. The RP proposes a concentration of 50 $\mu g\ kg^{-1}$ zinc during HFT (significantly higher than the 10 $\pm$ 5 $\mu g\ kg^{-1}$ specified for normal operation). The RP claims (based on the experimental data presented for Alloy 600 and stainless steels) the higher level of 50 $\mu g\ kg^{-1}$ zinc is optimal for oxide formation; increased concentrations resulting in thinner oxides and lower corrosion rates with minimal further benefits at higher concentrations. I am content with the proposed concentrations at this stage, noting that further justification for the proposed zinc concentration for HFT, taking into account any additional available OPEX, should be provided by the licensee; I consider this to be a minor shortfall.

533. The RP proposes the use of natural boron during commissioning, rather than EBA. While this has no direct consequence for the HFT process itself, all of the natural boron must be effectively removed from the plant before operation with nuclear fuel to mitigate any possibility of inadvertent boron dilution, which is noted to be the main chemistry aim of the final part of the non-active commissioning stage II (Ref. 121). The selection of natural boron was based on economic grounds and a full risk assessment for the selection of natural boron during HFT was not presented. Further justification by the licensee will strengthen the safety case in this area and I consider this to be a minor shortfall.

534. A number of support systems will be required during HFT, such as the RCV [CVCS] and REN [NSS] (assessed separately in sub-sections 4.2.2.1 and 4.2.8.1 respectively). Capability and availability of these systems was not considered as part of the GDA assessment, and can only be undertaken once fully developed HFT procedures are available, however as the suggested levels are stated to fall within the normal operating ranges for these systems, no concerns are expected in this area provided availability can be assured (Ref. 116). Whilst this is true for the majority of the chemistry controls, as noted above, the zinc concentration during HFT is approximately 5 times that of normal operating values. On the other hand, it is recognised that zinc may not be detectable during HFT due to rapid absorption into oxide surfaces. Further justification of the requirements and capability in this area in the site-specific stages will strengthen the safety case. I consider this to be a minor shortfall.

535. Sampling during HFT may prove an important step and it may be that analysis of chemistry during these periods could be used as a 'hold point' and as an indicator of HFT progress. Consideration of this type of control should be justified and included in a fully developed HFT procedure specific to UK HPR1000.

536. The RP recognises the importance of purification during HFT. Maximisation of the coolant clean-up rate will influence the benefits gained from any HFT process. The UK HPR1000 approach will use the main RCV [CVCS] demineralisers operating at a

continuous throughput at maximum possible flowrate for the periods of pre-oxidation and de-lithiation prior to acidification. The capability of the RCV [CVCS] in UK HPR1000 is assessed in sub-section 4.2.2.1 and was outside the scope for this assessment area. Even so, recognition of the need to maximise purification is encouraging.

537. I believe the use of zinc during HFT is a positive step. I support development and implementation of an UK HPR1000 HFT process, based upon the fundamental principles outlined by the RP for UK HPR1000. Post-GDA, a fully developed HFT procedure should be developed for UK HPR1000, building upon the evidence presented for GDA, including further knowledge and experience particularly from other HPR1000 units commissioned prior to any UK new build.

4.2.11.1 Strengths

538. The RP has considered a range of experimental data, relevant plant OPEX and RGP. The approach proposed appears reasonable and consistent with the current knowledge and RGP in this area.

539. I consider the RP has provided a good basis from which the licensee can build on, taking into account future OPEX (particularly of the hot functional testing chemistry regimes) which may be available before commissioning of the UK HPR1000.

4.2.11.2 Outcomes

540. Based upon my assessment of the commissioning of the UK HPR1000 described in sub-section 4.2.11, I have identified three minor shortfalls.

4.2.11.3 Conclusion

541. In the context of making a reasoned and informed choice, it is clear the RP has given significant consideration to the chemistry controls during commissioning, drawing on RGP. Evidence and assertions to support the basis of the conclusions reached are presented in the relevant topic report (Ref. 121). I am satisfied the RP has supplied sufficient information at this stage to justify what chemistry control during commissioning is aiming to achieve for UK HPR1000; this can be further developed as part of normal business by the licensee.

542. I have identified no Assessment Findings as a result of this part of my assessment.

543. The principal SAPs relevant to this part of my assessment are those related to chemistry (ECH.1), commissioning (ECM.1), ageing and degradation (EAD.1) and heat transport systems (EHT.5). I am content that, overall, the expectations set out in these assessment principles have been met by this part of the safety case.

4.3 Secondary Circuit

544. The primary objective of secondary side water chemistry control in a PWR is to minimise corrosion damage and performance losses for all secondary system components and thereby to maximise the safety, reliability and performance of the secondary systems. To achieve this objective, the water chemistry must be compatible with the many diverse systems that make up the secondary circuit. In particular, the SGs may not be replaceable or easily repaired and are a barrier to contain radioactivity; their degradation or performance loss greatly affects the overall plant performance and safety, especially where they form part of the pressure barrier to the primary circuit. A number of other degradation mechanisms throughout the secondary circuit are also influenced by the operating chemistry, such as FAC. The secondary coolant chemistry must therefore be balanced to account for all of these factors, in

addition to ensuring that other constraints or requirements (such as environmental discharge limits) are met.

4.3.1 Overview

545. The UK HPR1000 secondary circuit, like most PWRs, consists of a number of different systems constructed from a range of materials. The main equipment of the secondary circuit includes the SGs, steam turbine, Moisture Separator Reheaters (MSRs), condensers, condensate pumps, low pressure feedwater heaters, deaerator, feedwater pump and high-pressure feedwater heaters as shown in Figure 5.

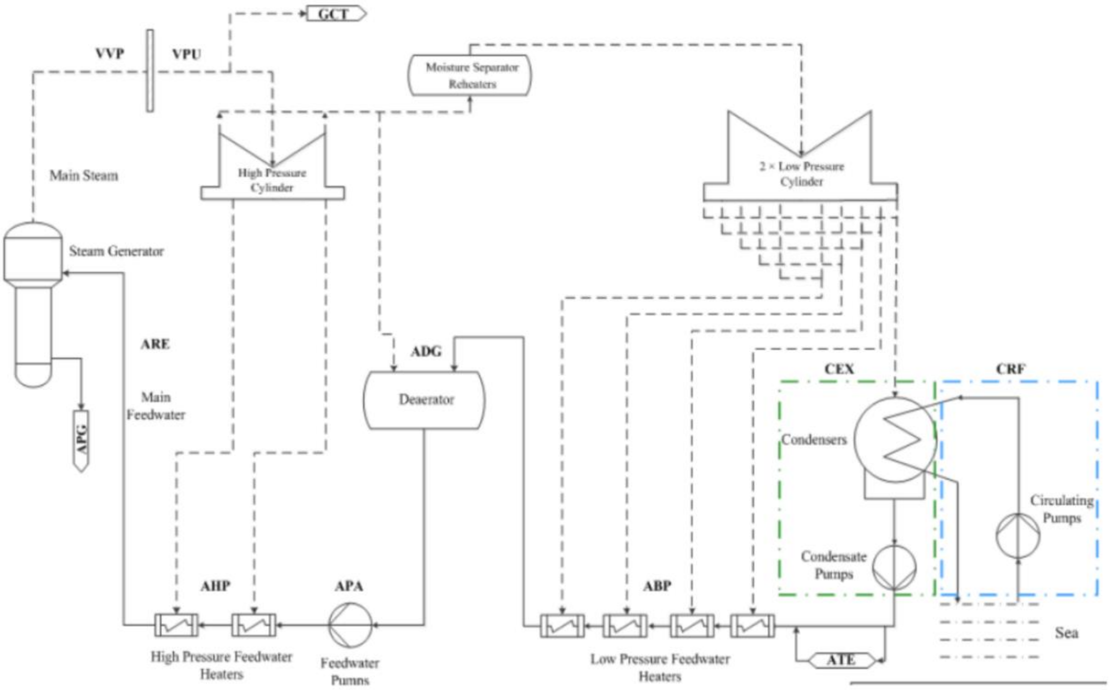

Figure 5: Simplified diagram of the steam and power conversion systems in UK HPR1000 **(Ref. 5)**

546. The bulk coolant in the SG is heated to saturation through heat transfer from the primary circuit and the resultant two-phase (steam-water) flow passes up the SG and is dried by the moisture separators and dryers in the top of the SG; the dried steam exits to the turbine.

547. Once within the High Pressure turbine, the steam temperature and pressure decreases and two-phase flow conditions form. As in all PWRs, the formation of wet steam conditions within the High Pressure turbine leaves all pipework and components that receive wet steam from the turbine potentially susceptible to corrosion, including FAC. Steam from the High Pressure turbine exhaust is dried and reheated in the MSRs to provide dry steam to the Low Pressure cylinders. The turbine exhaust steam is then condensed back to water in a condenser which is cooled by the Circulating Water System (CRF [CWS]). Steam may also be dumped directly from the SGs to the condenser in some circumstances (through the Turbine Bypass System (TBS [GCT]), which can also lead to a risk of condenser tube erosion. Control of oxygen ingress to the condensate is important at this stage.

548. The condensate is pumped up from the condenser hotwell by condensate pumps. The condensate then passes through the Low Pressure Feedwater Heater System (ABP [LPFHS]) and the Feedwater Deaerating Tank and Gas Stripper System (ADG [FDTGSS]). The pressure of the feedwater is then increased and it is further heated by

the High Pressure Feedwater Heater System (AHP [HPFHS]) before it is delivered back to the inlet of the SG.

549. The Start-up and Shutdown Feedwater System (AAD [SSFS]) supplies feedwater to the SGs bypassing the AHP [HPFHS] during plant start-up, hot standby and shutdown conditions.

550. The RP makes the following sub-claim on secondary circuit chemistry in UK HPR1000 (Ref. 5): “3.3.10.SC21.4 The secondary chemistry and process are optimised in all operating modes to maintain the integrity of the SG in the secondary side”. The main sub-claim is supported by a series of arguments and sub-arguments, including that the corrosion of the secondary circuit systems and the transport of CPs to the SGs is minimised SFAIRP through materials selection and chemistry control.

551. As part of my assessment of secondary circuit chemistry during GDA, I have sampled several interrelated areas which cover the main chemistry related safety concerns in the secondary circuit. In line with the other chemistry assessment topics, I reviewed the approach and expectations for secondary circuit chemistry in UK HPR1000 and the justification that the plant can safely support the anticipated chemistry. The areas sampled were:

- Materials, Design and Chemistry
- Materials Integrity and Corrosion
- Sampling Systems
- Start-up and Shutdown Chemistry

552. Each of these areas is discussed in detail in the following sections.

4.3.2 Materials and Design

553. The secondary circuits of power plants are often made up of multiple systems containing many different material types. It is therefore necessary, in line with SAP ECH.2, to reconcile the often divergent requirements of distinct systems to achieve an appropriate overall balance for safety, both for the chemistry parameters and the design (Ref. 2). In order to define an appropriate overall chemistry regime, a thorough understanding of the different system designs and materials is required. The following sections are focussed on the materials and design choices for the main chemistry-related systems in the UK HPR1000 secondary circuit.

4.3.2.1 Steam Generators (SGs)

554. The SG heat transfer u-tubes are the main interface between the primary and secondary circuits of a PWR. The secondary circuit feed water is in permanent contact with the outer diameter of the u-tubes, where it absorbs the heat to create steam. The steam quality is improved by the two moisture separators located in the upper section of each SG.

555. The secondary circuit components and design of the SGs are described in Section 5.2 of PCSR Chapter 21 (Ref. 5). The major assemblies in the SG are composed of a mixture of stainless steel, LAS and nickel-based alloys (detailed in Table 11).

Table 11: Secondary Side Materials of the SG Major Components

Material group	Material and Code	Equipment and Component
LAS	ASME SA-508 Gr.3 Cl.2 (Cr content 0.1-0.25%)	Secondary shell cans (including feedwater nozzles)
		Shell cone
		Steam Drum Head (including steam outlet nozzle)
		Tubesheet
	ASME SA-335 P22, SA-234 WP2 (Cr content 1.9-2.6%)	Feedwater assemblies
Stainless Steel	ASME SA-240 Type 410S	Tube support plates
	ASTM A 511 MT304L ASME SA-240 Type 304L	Separator
Nickel-based alloy	ASME SB-167 UNS N06690	Main feedwater header J-tube assembly
	ASME SB-163 UNS N06690	Tubes

556. My assessment concentrated on the design features and material choices, and how they influence the corrosion performance and plant chemistry. This is the area where OPEX of older designs has noted a number of weaknesses that result in degradation, which I will discuss in my assessment below.

557. The RP makes several sub-arguments relating to the material selection of the SGs, including that corrosion resistant materials are selected for the SG tubes and internals and that detrimental impurities, such as lead and copper, are limited by optimised material selection to minimise pitting and SCC of SG tubes.

558. The main reports detailed in the PCSR are the 'Material Selection Summary Report', Ageing and Degradation Justification Summary Report (Ref. 48, Ref. 57). I have also reviewed the following RP submissions as part of my assessment:

- 'Material Selection Report of SG' (Ref. 56)
- 'Ageing and Degradation of SG' (Ref. 59)
- 'Steam Generator Component Safety Report' (Ref. 123)
- 'Design Specification of Steam Generator' (Ref. 124)

559. As part of my assessment, I raised one RQ, RQ-UKHPR1000-1640, to gain further clarification and evidence to support the claim and argument made in the PCSR (Ref. 3).

560. The RP identified the main ageing and degradation effects which are of concern for the SG which can be influenced by coolant chemistry, as shown in Table 12.

Table 12: Ageing Effects and Mechanisms of the Steam Generators Influenced by Secondary Coolant Chemistry

Ageing and Degradation Effect	Related Mechanisms
Material Loss	Crevice corrosion, Denting, Erosion, FAC, Galvanic corrosion, General corrosion, Pitting corrosion
Cracking	IGA, SCC

561. The most significant material choice made for the UK HPR1000 SGs is the use of Alloy 690TT for the SG tubes. I consider this to be RGP; almost the standard material of choice for SG tubing material and is consistent with the latest replacement SGs. Protection of the SG tube material is a determining factor in any chemistry regime. While accepted that, overall, this material has a superior corrosion behaviour than alternatives, the RP recognises a number of environments still pose risks to Alloy 690TT, and assessment of this aspect is detailed in sub-section 4.2.4.1 (Ref. 32, Ref. 59). The thermal treatment, avoidance of cold work and cleanliness control during manufacture, installation and commissioning is also important to the corrosion propensity of the SG tubes in the secondary coolant; this is no different to that of the primary coolant and is discussed in sub-section 4.2.4.1.

562. The SG tubes are an inverted u-type bundle which pass through the tubesheet in a triangular pitch arrangement. The inverted ends are welded to the primary side of the tubesheet cladding (Alloy 52, the corresponding weld material for Alloy 690TT). After the welds pass a helium leak test, the tubes are full depth expanded to the tubesheet, to just below the secondary side, to minimise crevices. The RP claim tube expansion equipment and procedures have been demonstrated to minimise residual stresses in the transition from the expanded to unexpanded zone (Ref. 123).

563. Further details of the expansion process, residual stress and the impact on SG integrity was requested in RQ-UKHPR1000-1640. The RP responded that a verified expansion process would be controlled to length, pressure and holding time, and controls to ensure cleanliness of the tubesheet holes were in place. No detail was provided concerning expected crevice size, only that it was "minimised". The RP stated that the maximum and average residual stress on the outer surface of the expansion transition was <50% and <25% tube yield strength respectively. The RP also note that "tube expansion equipment and procedure have been demonstrated to minimise residual stresses in the transition from the expanded zone to the unexpanded zone. Measures are taken to assure that tubes are expanded close to but just below the secondary side of the tubesheet" (Ref. 123). I consider this to be an acceptable response at this stage; the manufacturing methods are in line with RGP, and manufacture specifications will be finalised in the site-specific stages. These specifications should include details of the allowable residual stress and crevice size, and I consider this to be part of normal business for the licensee.

564. The SG tubes are supported by nine tube support plates, spaced over the bundle height which consist of a series of anti-vibration bar assemblies positioned between each layer of tubes. The lattice structure of the tube support plates is claimed to make the "flow velocity high enough" between the tube and support plate so that the accumulation of impurities and CPs is reduced. I requested further details to support this claim in RQ-UKHPR1000-1640. The RP responded that the lattice grid structure had a larger clearance flow area and smaller resistance than a broached support plate design, with a circulation ratio (defined as "the ratio between the total flow rate

circulating in the riser and the steam flowrate at the outlet of the SG") at the FCG3 reference plant of ~ 5. I was satisfied with the response provided; it is supported by OPEX of Babcock Wilcox recirculating and once through PWR SGs (Ref. 125, Ref. 126).

565. The tube support plates and anti-vibration bars are fabricated from a martensitic stainless steel (Type 410S, Table 11). This high chromium steel should provide adequate corrosion resistance within the secondary coolant, provided an appropriate chemistry regime is maintained.

566. The SG shell is formed by a conical shell, cylindrical shells and an elliptical steam drum head. The tubesheet is fabricated from the same material, as an integrated forged piece. The SG shell also includes a number of manways, hand holes, instrument nozzles, and main feedwater, auxiliary feedwater and blowdown nozzles. All these components are fabricated from SA-508 Gr.3 Cl.2 steel, with a minimum specified chromium content of 0.1% for protection against FAC (discussed further in sub-section 4.3.4.2).

567. The UK HPR1000 has a two feedwater nozzle and system; main and auxiliary. The auxiliary system will supplement the main system during certain operating conditions and transient events. The main assemblies are fabricated by with a LAS (SA-335 P22 and SA-234 WP2). The J-tube assembly and main feedwater header are fabricated from Alloy 690TT. The J-tube prevents draining and directs feedwater downwards in the downcomer towards the tube bundle region. Perforations within the J-tube are sized to provide "debris-filtering functionality" and help to prevent loose parts.

568. The SG design includes a dedicated sludge collector located on the primary moisture separator support deck, which is designed to separate floating particles in the secondary coolant. I asked for further information concerning its efficiency in RQ-UKHPR1000-1640. The RP responded that Computational Fluid Dynamic (CFD) analysis predicted removal of 30-50% of the sludge transported to the SG with the feedwater, whilst some US OPEX suggests a potential for removal in excess of 50%. Full evidence (detailed descriptions and analysis) was not provided during GDA, and justification will be required by the licensee during the site-specific stages as part of normal business.

569. Despite design improvements and chemistry control, the accumulation of some sludge and deposits within an operating SG is inevitable (Ref. 127). An important consideration then becomes the provision in the design for inspection and cleaning (lancing), especially in low flow areas. Sludge accumulation and deposition continues to cause operational issues in some plants (Ref. 128). RQ-UKHPR1000-1640 queried a number of aspects related to provisions for sludge removal and anticipated quantities. The RP responded as follows:

- The design incorporates two and four circumferentially equispaced handholes just above the top tube support plate and tubesheet secondary surface respectively. These allow inspection, instrument access and provision for cleaning operations.
- Flush tubes and drain pipes are designed for the high pressure water lancing and suction to remove sludge from the sludge collector.
- Recommendations are made for cleaning at each maintenance outage or every two years. However, the cleaning criteria will be written by the licensee in the site-specific stages.
- CGN OPEX of an average sludge-lanced deposit from one SG tubesheet during maintenance outages indicates it to be of the order of "only 2 kg".

570. Further analysis and details concerning the efficiency of the sludge collector, likely locations and quantities of CP will be required during the site-specific stages as part of normal business.

571. The corrosion allowance of the SG design, in addition to general corrosion and FAC, includes an allowance for hydrochemical cleaning, which I consider to be a positive consideration in the design at GDA stage. The adequacy of this claim is not part of GDA and would require a justification by the licensee as part of normal business as it depends on the detailed proposal and conditions at that time.

4.3.2.2 Steam Generator Blowdown System

572. The Steam Generator Blowdown System (APG [SGBS]) is used to maintain the quality of the water/steam mixture within the secondary circuit by continuously removing a proportion of the feedwater flow from the SG. The APG [SGBS] is described in PCSR Chapter 21 and in the related secondary circuit DSR (Ref. 5, Ref. 129).

573. During normal operations the maximum blowdown flow rate of the APG [SGBS] is 1% of the rated steam flow rate, which is approximately 63 $t\ h^{-1}$ (although the expected flow rate range in normal operations is between 10 and 63 $t\ h^{-1}$). The blowdown water is cooled to below 56 °C by the regenerative heat exchanger, which is cooled by the Condensate Extraction System (CEX [CES]), and then depressurised by the blowdown control valve before being discharged into the APG [SGBS] treatment unit. The treatment unit consists of two 100% capacity filters in parallel (only one filter is in operation at any one time), and two 50% capacity demineralisation trains in parallel. The two demineralisation trains are both in operation under normal operating conditions and each contain a separate cation and anion ion exchange vessel and a resin trap. The treated blowdown water is sent to the condenser of the CEX [CES] for reuse. The design features no provision for the regeneration of exhausted ion exchange resins; this results in some restrictions on the operating chemistry, as described in sub-section 4.3.3.

574. When the demineraliser ion exchange resins are spent they are flushed to the low activity resin separation tank of the Solid Waste Treatment System (TES [SWTS]) using demineralised water from the NI Demineralised Water Distribution System (SED [DWDS (NI)]).

575. Justification for the sizing of the demineraliser ion exchange beds is provided in the DSR, with reference to resin usage life data from CPR1000 plants, which operate with a similar APG [SGBS] treatment unit design (Ref. 129). The DSR draws a detailed comparison with the CPR1000 blowdown water chemistry control requirements and APG [SGBS] demineraliser capacity in order to justify the UK HPR1000 APG [SGBS] design. However, the justification lacks supporting plant data from CPR1000 which would further substantiate the ability of the system to remove specific impurities to the required levels; I consider this to be a minor shortfall. The anion resin bed is designed to operate for 18 months prior to resin replacement, and the cation bed for 6 months. In response to RQ-UKHPR1000-1060, the RP provided details of the resin replacement process; during resin changes the system capacity is halved (Ref. 3).

576. The main components of the APG [SGBS] treatment system, such as the demineraliser vessels, filters, the tube side of the regenerative heat exchanger and the piping, are constructed from stainless steel in order to reduce the potential for corrosion in these potentially susceptible areas of the system. Other APG [SGBS] components are made from carbon steel, including the blowdown isolation valves and the shell side of the regenerative heat exchanger. Local conditions within the APG [SGBS] leave some components susceptible to FAC. Sub-section 4.3.4.2 discusses the RP's approach to evaluating and controlling FAC risk, including for the APG [SGBS].

577. Overall, the design and materials choices for the APG [SGBS] appear reasonable, and an adequate justification of the clean-up capacity of the system has been provided.

4.3.2.3 Main Feedwater Systems

578. The feedwater systems are described in PCSR Chapters 21 and 11 and serve to transfer condensate from the condenser to the SG inlet (Ref. 5, Ref. 22). The condensate first passes through the ABP [LPFHS] and the ADG [FDTGSS]. Its pressure is then increased, and the feedwater is further heated by the AHP [HPFHS] before it is delivered to the SG inlet. The AAD [SSFS] supplies feedwater to the SGs bypassing the AHP [HPFHS] during plant start up, hot standby and shutdown conditions. In order to maintain a reducing environment in the SG secondary side and to minimise pitting and SCC of SG tubes, the dissolved oxygen concentration is controlled during both normal operation and commissioning; the bulk of the oxygen is removed by the deaerators of the ADG [FDTGSS].

579. The PCSR describes materials choices for the key feedwater systems. Feedwater piping uses LAS (with 1-1.25 wt% chromium) and feedwater heater piping is made from stainless steel. A Materials Selection Report describing the optioneering carried out to select an appropriate material for the Main Feedwater Line (MFL) component of the Main Feedwater Flow Control System (ARE [MFFCS]) was also provided (Ref. 130). The primary purpose of the MFS is to control the flow of feedwater to the SGs (temperature, pressure and flow rate), ensuring the SG level remains within the allowable tolerance range. The report captures the susceptibility of the component to FAC and other relevant degradation mechanisms and draws on relevant OPEX from similar designs. P280GH, a carbon steel with minimum 0.2% chromium, is selected with appropriate consideration given to compatibility with the operating environment and avoidance of reliance on chemistry control. From a Chemistry perspective, I am content with the material selected and the justification set out in the safety case.

4.3.2.4 Condensate Systems

580. Exhaust steam from the Low-Pressure turbine cylinders, or directly from the SGs, is condensed back to water in a condenser which is cooled by the CRF [CWS]. The condensate is pumped up from the condenser hotwell by condensate pumps, after which the Condensate Polishing System (ATE [CPS]) purifies the condensate to meet the secondary water chemistry specifications.

581. The ATE [CPS] is described in PCSR Chapter 21 and a DSR (Ref. 5, Ref. 129). The system is designed to remove impurities from the condensate, using filters and ion exchange beds, to ensure the feedwater and steam meet the relevant quality specifications. Whilst the APG [SGBS] controls the feedwater quality during normal operations, leakage of circulating water into the condenser can lead to high levels of impurities being introduced to the condensate, which the APG [SGBS] does not provide sufficient clean-up capacity to remove. The ATE [CPS] is therefore put into operation in the event of condenser leakage to remove impurities from the totality of the secondary circuit condensate and provide additional time for unit maintenance or shutdown when serious condenser leakage is detected. The system is also put into operation during start-up of the unit to remove species generated during installation or maintenance. Demineralised water is used to flush the secondary circuit until an acceptable water quality is achieved; the flushing water is purified by the ATE [CPS], thus decreasing the time and volume of water required for the start-up of the unit.

582. The polishing unit of the ATE [CPS] consists of five 25% capacity cation demineralisers and five 25% capacity mixed bed demineralisers, each with an outlet resin trap to prevent resin leakage into the feedwater. During operation, four cation beds and four mixed beds are in service with the remaining beds in standby. As discussed in sub-section 4.3.3, the UK HPR1000 will operate with a High-All Volatile Treatment (H-AVT)

secondary circuit chemistry regime, with ammonia dosing. Since the cation resin selectivity for ammonia ions in water is higher than that for sodium, the ATE [CPS] uses a cation bed to first remove ammonia ions, followed by a mixed bed to remove the remaining impurities.

583. Due to the high pH and the presence of ammonia in the coolant, the cation bed resin is exhausted in a relatively short period. Provision is made for resin regeneration in an auxiliary unit consisting of separate regeneration and storage tanks for the cation resin and the mixed bed resins, provision for addition of acid/alkali regenerants and for resin washing with demineralised water. The cation bed resin is designed to be regenerated approximately every 64 hours, and the mixed bed resin every seven to ten days during operating with condenser leakage of less than 60 L/hr of seawater. Justification for the sizing of the ion exchange beds is provided in the DSR, with reference to resin usage life data from CPR1000 plants, which operate with a similar ATE [CPS] design (Ref. 129). Whilst some limited OPEX from the use of the ATE [CPS] during condenser leakages in CPR1000 plant is presented in the DSR, the justification lacks detailed supporting data or other information to show that the chemistry can be maintained during condenser leakage and how much leakage may be tolerable, from a chemistry perspective, before plant shutdown is required (Ref. 129). This justification will need to be reviewed and developed by the licensee as the relevant operating limits and conditions necessary in the interests of safety are developed; I consider this to be a minor shortfall.

584. The ion exchange beds and resin regeneration and storage tanks are constructed from rubber lined carbon steel; the RP notes that this is to minimise corrosion caused by interaction with the acidic and alkaline media, however the justification for this choice of material over other options could be strengthened in the safety case. I consider this to be a minor shortfall. The system piping is largely made from stainless steel.

585. As described in sub-section 4.3.5.1, online sampling is performed by the SIT [FCSS] to detect condenser leakage and monitor impurity ingress from make-up water and chemical dosing. Continuous monitoring of feedwater sodium and cation conductivity is also performed during start-up.

586. The design and materials choices for the ATE [CPS] appear reasonable, and whilst I have raised minor shortfalls relating to supporting evidence that the system can maintain the required chemistry during a condenser leak and the justification for materials choices, I consider that overall an adequate case has been presented for GDA.

4.3.2.5 Steam Systems

587. The Main Steam System (MSS) is designed to transfer steam from the SGs to the turbines and other steam consuming systems during power operation, as described in PCSR Chapter 11 and the relevant SDM (Ref. 22, Ref. 131).

588. In UK HPR1000, the MSS is divided into 3 trains, each connected to a SG steam nozzle and consisting of a Main Steam Isolation Valve, Main Steam Safety Valve, and Main Steam Line (MSL). A material selection report describing the optioneering carried out to select an appropriate material for the MSL was also provided (Ref. 132). The report captures the susceptibility of the component to general corrosion, FAC and other relevant degradation mechanisms and draws on relevant OPEX from similar designs. P280GH carbon steel is selected, with appropriate consideration given to compatibility with the operating environment and the secondary circuit chemistry regime (as described in sub-section 4.3.3). From a Chemistry perspective, I am content with the material selected and the justification provided.

4.3.2.6 Chemical Dosing Systems

589. The secondary circuit coolant is dosed with chemicals during normal operations in order to control certain chemistry parameters to the required levels. The UK HPR1000 achieves this via a Chemical Reagents Injection System (SIR [CRIS]) which is described in PCSR Chapter 21 and in an associated DSR (Ref. 5, Ref. 129).

590. To fulfil the chemical dosing requirements for the condensate and feedwater, ammonia and hydrazine are injected via a number of dosing points including at:

- The condensate header, to counter the loss of ammonia and hydrazine in the APG [SGBS] purification system, from bled steam of CVS, and from the ATE [CPS] purification function (during start-up or condenser leakage).
- The deaerator downcomers and recirculating lines, to make up for the loss of ammonia and hydrazine from oxygen removal in the deaerator.
- The SG feed lines from the ASG [EFWS] via the dosing lines of the NI Chemical Reagents Distribution System (SIH [CRDS (NI)]) for conditioning of the SG water during SG wet lay-up.

591. Ammonia and hydrazine dosing to the condensate dosing points is automatically controlled through continuous monitoring of condensate and feedwater chemical parameters by the SIT [FCSS]. Other dosing points are manually dosed based on SIT [FCSS] measurements.

592. Tri-Sodium Phosphate (TSP) is also introduced into the closed loop cooling systems via the SIR [CRIS]. This part of the system consists of two sets of dosing devices located within the turbine hall, with stainless steel pipework running from the dosing devices to the relevant dosing points. The parts of the SIR [CRIS] delivering TSP are sufficiently segregated from the ammonia and hydrazine injection equipment so as to aid in preventing inadvertent dosing of the incorrect chemical into the secondary circuit.

593. Overall, the design of the SIR [CRIS] appears reasonable, consisting of a simple set of storage and mixing tanks, pumps and injection lines. Due to the corrosive nature of the chemicals stored and delivered by the SIR [CRIS], the ammonia and hydrazine metering tanks and the system pipework and valves are all, appropriately, constructed from austenitic stainless steel.

4.3.2.7 Demineralised Water Systems

594. PCSR Chapter 21 briefly describes the UK HPR1000 demineralised water distribution systems (Ref. 5). Demineralised water is required for many of the NI and conventional island systems in a PWR, and it is often produced on site in a water treatment plant which filters and purifies towns water to a condition suitable for use by the plant systems. UK HPR1000 has a SED [DWDS NI] and a SER [DWDS CI]. Demineralised water treatment systems can often be site specific due to the different feedwater supply compositions. As such I have not performed a detailed assessment of this system during GDA, however I note that there is no reason to suggest that a suitable system could not be specified at the site-specific stage.

4.3.2.8 Strengths

595. Overall, the RP has provided a coherent, adequate safety case to justify the chemistry functions of the relevant secondary circuit systems, at an appropriate level within the safety case.

596. The design of the SGs in the UK HPR1000 incorporate a number of features which are based on RGP. These include selection of corrosion resistant materials, full depth

hydraulic expansion technology, lattice design tube support plates and sludge collector.

4.3.2.9 Outcomes

597. Based upon the assessment of secondary circuit design and materials in UK HPR1000 described in sub-sections 4.3.2.1 to 4.3.2.7 above, I identified three minor shortfalls.

4.3.2.10 Conclusions

598. Based on the outcome of my assessment of secondary circuit materials and design in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA. In general, I consider that that adequate evidence has been presented to support the argument that that the corrosion of the secondary circuit systems, and the transport of CPs to the SGs, has been minimised SFAIRP.

599. The principal SAPs relevant to this part of my assessment are ECH.1 to ECH.3, EAD.1, EMC.13 and EMC.16. I am content that, overall, the expectations set out in these assessment principles have been met by this part of the safety case.

4.3.3 Chemistry Control

600. It is necessary to consider the differing requirements of the individual systems when specifying a secondary circuit chemistry regime, in order to achieve an appropriate overall balance for safety. A number of factors need to be taken into account, including control of corrosion and deposition, materials compatibility and compliance with environmental discharge requirements. PWR secondary circuit chemistries have evolved with advances in materials and design, and with lessons learned over time. At present a typical PWR secondary circuit operating chemistry is likely to be based on:

- Addition of a volatile pH control agent
- Minimisation of feedwater impurity levels
- Maintaining a reducing environment in the SGs through hydrazine addition

601. PCSR Chapter 21 and a ‘Topic Report on Power Operation Chemistry’ describe the secondary circuit chemistry regime for UK HPR1000 as AVT with ammonia for pH control and addition of hydrazine as an oxygen scavenger and reducing agent, in order to maintain reducing conditions and impurity controls (Ref. 5, Ref. 28). The limits during power operation are set out in Table 13. The general regime suggested is reasonable and in alignment with other operating commercial PWRs worldwide, as well as with the recommendations made in international guidelines.

Table 13: Secondary Circuit Water Chemistry Limits during Power Operation **(Ref. 30)**

System	Control Parameter	Units	Operating Window	Limit
High Pressure Feedwater Heater System / Main Feedwater Flow Control System	$pH_{25\,°C}$		[redacted]	[redacted]
	Oxygen	$\mu g\ kg^{-1}$	[redacted]	
Steam Generator Blowdown System	Cation conductivity (25 °C)	$\mu S\ cm^{-1}$	[redacted](1)	
			[redacted](2)	[redacted](2)
	Sodium	$\mu g\ kg^{-1}$	[redacted](1)	
			[redacted](2)	[redacted](2)

(1) Power ≤25% (2) Power >25%

4.3.3.1 pH Control

602. An optioneering study was carried out to select and to justify the preferred alkaline agent for pH optimisation; this is presented in 'Application Analysis of Alkaline Agent in the Secondary Circuit' (Ref. 133).

603. The optioneering report notes that the main objectives of controlling the pH are to minimise (Ref. 133):

- general corrosion and FAC in the secondary circuit systems, especially FAC in steam systems with two-phase flow; and
- SG corrosion, particularly the SG tubes but also the tube support plates and the tube sheet.

604. These objectives are linked to the RP's sub-argument 3.3.10. SC21.4-A2.2: AVT is applied via the SIR to control the pH and minimise general corrosion, FAC and to limit transport of CPs to the SGs.

605. The 'Topic Report on Power Operation Chemistry' notes that minimisation of transfer of CPs to the SGs, and mitigation of FAC risks to carbon steel and LAS components, are also objectives of feedwater pH control in UK HPR1000 (Ref. 28). FAC risk is assessed in more detail in sub-section 4.3.4.2, although its management is very much related to the choice of secondary circuit chemistry.

606. As shown in Table 13 and in Annex 3, the RP has specified a target operating $pH_{25\,°C}$ range of [redacted], with limits at [redacted]. In justifying these limits, the RP notes that small changes in pH can have a significant effect on the solubility of magnetite, and consequently on the rate of FAC of secondary circuit materials. The FAC rate decreases rapidly with increasing pH, although the dependence is much less when the $pH_{25\,°C}$ is greater than [redacted]. The RP also argues that, according to OPEX from its operating PWRs, the cation resin in the APG [SGBS] demineralisers will be rapidly exhausted when the feedwater $pH_{25\,°C}$ is greater than [redacted], which results in unnecessary disposal of chemical waste and radioactive solid waste.

607. Regarding the choice of pH raising agent, the RP presents three options – ammonia, morpholine (C_4H_9NO) and ethanolamine (C_2H_7NO) – in 'Application Analysis of Alkaline Agent in the Secondary Circuit' (Ref. 133). The conclusion of the optioneering

study performed is that all three options are compatible with the materials in the secondary circuit and that they can all satisfactorily suppress the risk of general corrosion and FAC; however, ammonia is selected as the preferred option at the GDA stage, on the basis that the RP has extensive experience with ammonia in its operating plants.

608. The RP recognises in the optioneering report that amines such as morpholine and ethanolamine are more effective at protecting two-phase flow areas (such as the MSR) from FAC and that OPEX, including from the EDF fleet, indicates that amine treatment has been successful in many PWRs. However, it notes that amine decomposition to organic acids is an issue, arguing that although there is minimal risk to material corrosion, the presence of organic acids increases cation conductivity at the APG [SGBS] and therefore leads to a lower sensitivity of detection of other anionic impurities, such as chloride and sulphate.

609. The RP notes that whilst ammonia is less effective at protecting areas of the circuit experiencing two-phase flow, by employing a H-AVT regime where the $pH_{25\,°C}$ is greater than [redacted], it can still effectively inhibit FAC by providing a higher pH_T within two-phase flow areas. The RP presents limited OPEX from German PWRs in support of H-AVT with ammonia, but also presents more detailed data from in-service testing of a H-AVT ammonia regime in a CPR1000 plant (Ref. 133). Samples were taken from a series of FAC sensitive areas to monitor the pH distribution around the circuit and the iron corrosion product content in the feedwater. The test results show a significant increase in pH at all of the areas monitored.

610. The RP recognises that the properties of different alkaline agents directly influence the exchange capacity of the ATE [CPS] cation resin and the system operating mode (Ref. 133). The ATE [CPS] (described in sub-section 4.3.2.4) cannot operate continuously during normal operation with H-AVT due to the high concentration of ammonia in the condensate, which would lead to rapid exhaustion of the resins. The APG [SGBS] also has a finite purification capacity, and once exhausted, the ion exchange resins will no longer remove impurities. Ammonia and ethanolamine will exchange with the cation resin in the system, resulting in shortening of the operating cycles of the cation demineralisers, which leads to an increase in the production of solid waste. For morpholine however, the operation cycle can be longer.

611. The RP has taken the positive step of removing copper sources from the UK HPR1000 secondary circuit. The exclusion of copper and copper alloys, as well as being consistent with OEF, means that restrictions on high pH regimes, which would be needed to mitigate copper alloy corrosion, are not necessary in UK HPR1000.

612. Overall, I consider that the RP has made an appropriate justification for its preferred choice of alkaline agent (ammonia, H-AVT) and operating limits for secondary circuit pH control. In doing so, the RP has given appropriate consideration to system materials choices, operating conditions, waste generation, corrosion threats and OPEX from other PWRs to reach a balanced regime. However, as some secondary circuit systems are outside of the RP's scope for GDA and, as such, materials choices have yet to be confirmed, the licensee will need to review and further justify the secondary circuit chemistry as the design matures, to ensure that the chemistry remains appropriate and that relevant risks are reduced SFAIRP. Based on the information provided during GDA, I consider that this can be achieved as part of the normal business activities of the licensee.

4.3.3.2 Oxygen Control

613. UK HPR1000 will operate with hydrazine addition to control oxygen and to produce a reducing environment in the secondary circuit. The feedwater deaerator is also used to remove bulk oxygen. In the 'Topic Report on Power Operation Chemistry', the RP

notes that the main objectives of secondary circuit oxygen concentration control during normal operation are to reduce the risk of IGA/SCC and pitting and crevice corrosion of SG tubes, and to minimise the general corrosion of the secondary circuit surfaces (and therefore minimise the transport of CPs to the SGs). Hydrazine can also thermally decompose into ammonia, hydrogen, and nitrogen, which contributes to the feedwater pH.

614. The RP proposes to operate with hydrazine addition in the range of [redacted] µg kg^{-1} (at least [redacted] times the oxygen concentration) and notes that this range is consistent with CPR1000 and HPR1000 (FCG3) units. Hydrazine is widely used in operating PWRs and the values suggested are reasonable, although potentially higher than may be necessary with a feedwater deaerator. I am satisfied with the arguments and evidence presented for oxygen control.

4.3.3.3 Impurity Control

615. In 'Topic Report on Power Operation Chemistry', the RP identifies the main sources of impurity ingress into the secondary circuit during power operations as (Ref. 28):

- Seawater ingress due to condenser leakage (sodium, chloride, fluoride, sulphate, calcium, magnesium.
- Ion exchange resin leakage into the secondary circuit (sulphate).
- Contamination of chemicals dosed into the secondary coolant including ammonia and hydrazine (sodium, chloride, sulphate, phosphate).
- Contamination of secondary circuit make-up water (sodium, silica).

616. As noted in Table 13, the control parameters proposed by the RP relating to impurity control are cation conductivity and sodium. The RP argues that, during power operation, chloride, fluoride and sulphate concentrations are very low and therefore difficult to measure online, adding that the conductivity measurement is more sensitive and is the preferred control measurement. The topic report explains that when an abnormal increase in cation conductivity is detected, chloride, sulphate and silica should be monitored and analysed (via grab sampling) in order to identify the source of impurity ingress in a timely manner. As I note in my assessment of Secondary Circuit Sampling Systems in sub-section 4.3.5, the ability to determine anion composition is common in the sampling and monitoring systems of other nuclear power plants. The sampling system needs to be able to provide information regarding anionic changes as this will affect corrosion mechanisms. Currently the RP's approach is to sample anions via grab sampling. Further justification should be provided at the site-specific stage to support the determination of anion composition via grab sampling, as opposed to online monitoring of species such as chloride, and I note in sub-section 4.3.5 that I consider this to be a minor shortfall.

617. The topic report explains that in the secondary circuit of a PWR, sodium is the main cause of IGA/SCC of SG (Ref. 28). Sodium can become concentrated within crevices inside SGs, leading to caustic conditions which may result in IGA/SCC of nickel-based alloys and stainless steels. The RP therefore proposes a control parameter limit on sodium concentration of up to [redacted] µg kg^{-1} when the unit is operating at above 25% of full power, based on OPEX from CPR1000 plants. This is broadly in line with international guidelines and I am satisfied with the arguments and evidence presented in this area.

618. The Secondary Water Chemistry Specifications also provide expected values for feedwater iron and chloride, sulphate, silica and suspended solids, which are diagnostic parameters monitored via grab sampling (Ref. 134). The values suggested appear reasonable and are consistent with operating plant experience.

4.3.3.4 Radiochemistry

619. The RP also proposes limits on ^{16}N and total gamma in the secondary circuit. During normal operation, fluid in the SG secondary side is non-radioactive. Therefore, radioactivity in the secondary coolant can be used as an indicator of SG tube leakage. The 'Topic Report on Power Operations Chemistry' states that the main aim of radiochemistry monitoring in the secondary circuit is to (Ref. 28):

- Monitor and detect any leakage of SG tubes, then to deal with it in a timely manner to avoid further deterioration of a leakage event.
- Prevent radioactive substances from spreading to the conventional island and the environment, and to minimise radiation exposure to personnel and the environment.

620. These objectives are linked to PCSR Chapter 21 argument 3.3.10. SC21.10-A1.8: Radiochemistry parameters are selected as the indicator for the integrity of safety barriers and radioactivity levels in the plant.

621. During a SG tube leak, the noble gases and ^{16}N tend to enter the MSL, whilst other radionuclides from the primary coolant will concentrate in the liquid phase at the secondary side of the SG. The topic report notes that, in line with common practices in operating PWRs, different indicators have been selected for the steam and liquid phases, to ensure timely, efficient monitoring of the integrity of the SGs:

- Steam phase – ^{16}N is selected to indicate SG leakage in the steam phase. It is generated from ^{16}O in the primary coolant, but its short half-life (about 7.13 s) means it can only be detected in the MSS, and according to OPEX from CPR, AP1000® and EPR™, the measurement is more reliable when the reactor power is above 20%. Total gamma of the MSS is therefore monitored to detect the leakage of SG when the power is below 20%.
- Liquid phase – Total gamma in the APG [SGBS] is selected to indicate SG leakage in the liquid phase, in line with practices at operating CPR1000 plants.

622. Both ^{16}N and total gamma are continuously monitored by the Plant Radiation Monitoring System (PRMS). Grab sampling for total gamma is also performed periodically via the REN [NSS] in order to verify the online monitoring result. The proposed control parameter limits are justified in the 'Radiochemistry Parameters Value' report and I am satisfied with the arguments and evidence presented in this area from a chemistry perspective (Ref. 31).

4.3.3.5 Strengths

623. The RP has provided a coherent, adequate safety case to justify the secondary circuit chemistry choices at an appropriate level within the safety case.

4.3.3.6 Outcomes

624. Based upon the assessment of secondary circuit chemistry in UK HPR1000 described in sub-section 4.3.3 above, I have not identified any Assessment Findings or minor shortfalls.

4.3.3.7 Conclusion

625. Based on the outcome of my assessment of secondary circuit chemistry in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA. I consider that adequate evidence has been presented

to support the RP’s argument that the selected chemistry regime can minimise general corrosion, FAC and to limit transport of CPs to the SGs.

626. The principal SAPs relevant to my assessment of the secondary circuit chemistry are ECH.1 to ECH.3. I am content that, overall, the expectations set out in these assessment principles have been met by this part of the generic safety case.

4.3.4 Material Integrity and Corrosion

627. The integrity of secondary circuit systems and components is an important consideration in specifying a secondary circuit chemistry regime. An unsuitable chemistry can result in deterioration and failure of components in severe cases, and at best, leads to an increase in the release of CPs which can accumulate in the SGs.

628. My assessment sampled the SG tubing to assess the impact of chemistry and materials on integrity. I also assessed the RP’s response to RO-UKHPR1000-0034, which was raised during Step 3 of GDA, on the RP’s approach to considering the risks of FAC which is a potential area of concern throughout the entire secondary circuit (Ref. 4).

4.3.4.1 Steam Generator Tube Integrity

629. SG tubes of a PWR are relatively thin walled in order to achieve good heat transfer; in UK HPR1000, the wall thickness is a nominal 1.02mm (Ref. 55, Ref. 56). The SG tubes in UK HPR1000 make up over 80% of the pressure boundary area of the primary circuit, and they are the main barrier between the primary and secondary circuits. Maintenance of their integrity is clearly important; loss of tube wall integrity will result in radioactivity release from the primary circuit into the secondary circuit, where it is potentially available for release to the atmosphere.

630. The RP makes several sub-arguments surrounding SG tube integrity, including that the corrosion of SG materials is minimised SFAIRP through material selection and chemistry control. The main chemistry related threats to the SG tube integrity are various localised corrosion mechanisms, including pitting corrosion, denting corrosion and Outer Diameter Stress Corrosion Cracking (ODSCC). Increased deposition in the SG of CPs originating from elsewhere in the secondary circuit (from mechanisms such as FAC) can increase the severity of these forms of corrosion as discussed below. FAC is assessed in sub-section 4.3.4.2.

631. The main reports detailed in the PCSR are the ‘Material Selection Summary Report’ (Ref. 48), ‘Ageing and Degradation Justification Summary Report’ (Ref. 57), ‘Topic Report on Impurity Control for the Operation’ (Ref. 32), and the ‘Design Substantiation Report on Associated Chemistry Control Systems: the Secondary circuit’ (Ref. 129). I have also reviewed ‘Material Selection Report of Steam Generators’ (Ref. 56) and ‘Ageing and Degradation of Steam Generators’ (Ref. 59) as part of my assessment.

632. The principal SAPs relevant to my assessment of the materials integrity are ECH.1 to ECH.3, EAD.1 to EAD.4, EMC.13 and EMC.16. I raised one RQ, RQ-UKHPR1000-1640, to gain further clarification and evidence to support the claim and argument made in the PCSR (Ref. 3).

633. My assessment of contamination control during fabrication (according to SAP EMC.16) and the SG materials in contact with the primary coolant and its associated ageing and degradation is covered in sub-section 4.2.4.1 of this report.

634. Pitting corrosion manifests as a number of small diameter cavities. It arises from a localised breakdown in surface passivity that does not immediately re-passivate, creating a concentrated area for corrosion. In most cases, whether a surface

immediately re-passivates or not depends on both chemistry and metallurgy factors. Susceptible locations may be areas of localised cold work, grain boundaries or carbides and other secondary phases in the metal (Ref. 135). From a chemistry perspective, it is generally accepted that pitting requires acidic, aggressive species (such as chloride or sulphate in the presence of an oxidant) and is accelerated by increasing temperature (Ref. 135, Ref. 136). Barriers to diffusion (for example, by scale or sludge) will enhance the chemical concentration and accelerate the pitting process. Based on the UK HPR1000 the following key elements will reduce the risk of pitting:

- Lattice design for the tube support plates will increase the flow velocity to minimise the accumulation of impurities and CPs (Ref. 59, Ref. 123).
- Sludge collector and drainage pipe reduce the CPs in the secondary circuit (Ref. 123).
- Consideration of controls for the concentration of impurities within the secondary circuit (Ref. 32).
- Removal of copper components within the secondary circuit to limit the copper impurities which can promote corrosion (Ref. 133).

635. I am content that the risks posed by pitting corrosion of the SG tubes in UK HPR1000 have been reduced SFAIRP.

636. Tube denting is the mechanical deformation, or constriction of the tube where it intersects the tube support plate or at the tube sheet. It arises from a build-up of CP deposits and/or excessive corrosion of the tube support plate tube sheet in the annulus at the intersection point that exerts a pressure on the tubes (since the oxide produced has a greater volume than the original metal). In extreme cases cracking has been observed in between the tube support holes and the secondary flow distribution holes causing distortions in the tube bundle to occur. More typically, even small dents can induce tensile stresses that lead to PWSCC and ODSCC/IGA, it may make the tubes more susceptible to high cycle fatigue and can give rise to inspection difficulties (Ref. 135).

637. The occurrence of denting is, in part, related to the inability of AVT chemistry regimes to maintain a buffering capacity, especially in the case of an ingress of aggressive species (such as chloride or sulphate in the presence of an oxidant). Based on the generic UK HPR1000 design, the same factors noted above to mitigate pitting corrosion together with following key elements will reduce the risk of denting:

- Full-depth hydraulic expansion of the tubes at the tube sheet to remove the crevices at this location (Ref. 55, Ref. 124).
- Material selection of a stainless steel for the tube support plates (Table 11) with an initial surface roughness Ra <6.3 µm to reduce the general corrosion rate (Ref. 56, Ref. 48).

638. I am content risks posed by denting corrosion of the SG tubes in UK HPR1000 have been reduced SFAIRP.

639. ODSCC includes both SCC and IGA and its occurrence is significantly impacted by the concentration of impurities (Ref. 135, Ref. 137). This has historically occurred in crevices, sludge pile regions, and locations with significant fuel deposit build-up (Ref. 135).

640. Based on the design of the UK HPR1000, the same factors noted above to mitigate pitting corrosion together with the following key elements will reduce the risk of ODSCC:

- Full-depth hydraulic expansion of the tubes at the tube sheet to remove the crevices at this location (Ref. 55, Ref. 124).

- Material selection of Alloy 690TT, with manufacture controls for cleanliness, stress relief to reduce residual stress and minimisation of cold work (Ref. 56, Ref. 59).

641. I am content that the risk posed by ODSCC of the SG tubes in UK HPR1000 has been reduced SFAIRP, provided an appropriate chemistry regime is maintained.

4.3.4.2 Flow Accelerated Corrosion

642. FAC refers to enhanced dissolution of the passive, and usually protective, oxide layers on the surface of carbon and LAS. It is a chemical effect that is primarily influenced by pH, hydrodynamics, material composition, oxygen concentration and temperature. FAC can occur in flowing water or wet steam, under single- or two-phase flow conditions and can be particularly prevalent under the conditions that can occur around the secondary circuit of a PWR. Not only can FAC lead to rapid failures of components, it is also implicated as a significant source of CPs transport around the secondary circuit. Material degradation by FAC can have serious safety consequences, with two events at PWRs having resulted in fatalities in the past (Ref. 138).

643. During Step 3 of GDA, the RP claimed that the risk of materials degradation due to FAC in UK HPR1000 had been minimised through materials choices and chemistry control. I was content that these were reasonable arguments during Step 3 of GDA, on the basis that further details of the analysis carried out and substantiation for these arguments would be required as GDA progressed. On this basis, I raised RQs (RQ-UKHPR1000-0457 and RQ-UKHPR1000-0561) to request details of the evidence supporting claims and arguments in this area, particularly on how control of FAC has been designed into the secondary circuit for susceptible areas and components (Ref. 3). The RP’s response to the RQs was insufficient to demonstrate effective protection of at-risk areas of the plant, nor did it identify any residual FAC risks that may need to be managed by the licensee. I therefore raised RO-UKHPR1000-0034 during Step 4 of the GDA, detailing the shortfalls identified against ONR’s expectations (Ref. 4).

644. In response to the RO, the RP produced a series of 3 reports (Ref. 139, Ref. 140, Ref. 141). The reports describe the RP’s:

- methodology for identifying the lines and components that are susceptible to FAC in UK HPR1000;
- review of secondary circuit lines and components against the methodology;
- identification and consideration of options available to mitigate or minimise the FAC risk SFAIRP for susceptible components; and
- identification of residual risks associated with FAC-susceptible components and any further actions that the licensee may need to carry out.

645. My assessment of the information provided by the RP is detailed in an assessment note ONR-NR-AN-21-022 (Ref. 142). I considered the matters raised in ONR-NR-AN-21-022 in this part of this assessment and I subsequently raised an Assessment Finding and a minor shortfall, as noted in the following sub-sections.

Approach to Evaluation of Flow Accelerated Corrosion Risk

646. The RP’s methodology for the evaluation of FAC risks in UK HPR1000, set out in ‘Approach for Evaluation of FAC Risks in UK HPR1000 Secondary Circuit’, involves ranking of components for their likelihood of failure due to FAC, based on material composition, flow conditions and environmental factors (pH, dissolved oxygen concentration and temperature) (Ref. 139). These criteria are applied to all secondary circuit systems containing water or steam and, initially, are used to screen out systems or portions of systems deemed not to be at risk of degradation due to FAC (for example, those systems or parts of systems constructed from stainless steel are not

considered further). The remaining systems are assigned a likelihood of failure level based on the criteria, and those components with significant likelihood of failure are considered further in ‘ALARP Report for FAC Risks in UK HPR1000 Secondary Circuit’, where options to reduce or eliminate FAC risk are presented (Ref. 140, Ref. 141). The RP also evaluates the consequence of failure of a sample of higher risk components in its report.

647. RO-UKHPR1000-0034 required the RP to justify its approach to evaluating FAC risks, taking into account all of the relevant influencing factors. To address this, the RP called upon an appropriate range of evidence to underpin its risk ranking criteria, including EPRI recommendations and experimental data from the literature. Additionally, the RP considered OPEX and lessons learned from a range of FAC-related events that have occurred worldwide, drawing on sources including the Nuclear Energy Agency (NEA) Component Operational Experience, Degradation and Ageing Programme (CODAP) and US Nuclear Regulatory Commission (NRC) bulletins.

648. The initial revision of Approach for Evaluation of FAC Risks in UK HPR1000 Secondary Circuit did not set out any intention to consider conventional health and safety risks to operators resulting from failures of components due to FAC in the RP’s consequence of failure ranking (Ref. 139). Instead, the consequence ranking was based solely on the SSCs safety classification and the impact of the SSCs failure on plant performance. Given the OPEX from FAC-related events at other PWRs that have involved fatalities or injuries to plant personnel as a result of fluid/steam releases from failed SSCs, I considered this to be a gap in the case. I raised RQ-UKHPR1000-1217 to query this position and the RP revised its methodology to give consideration to risks to operators for a small sample of SSCs where the maturity of the generic design was such that relevant information on plant layout and occupancy was available (Ref. 3). Whilst I consider this to be an acceptable approach for GDA, evidence to support the claim that the conventional health and safety risks associated with failures of SSCs due to FAC have been reduced SFAIRP for systems not considered at GDA should be provided as the necessary detailed design information becomes available. I consider this to be a minor shortfall.

649. The second ROA of RO-UKHPR1000-0034 also required the RP to identify the lines and components in the UK HPR1000 secondary circuit that are considered susceptible to material degradation by FAC, based on the methodology that it had set out. All water or steam-based secondary circuit SSCs have been considered by the RP in ‘Assessment Report for FAC Risks in UK HPR1000 Secondary Circuit’ (Ref. 140); this is a total of 20 systems. Out of the 20 systems considered, 11 were excluded entirely from further analysis during GDA on the basis of the chromium content of the material exceeding a 1.25% threshold (above which the likelihood of FAC occurring is very low), the system containing dry steam, or the system operating at a temperature outside of a range in which FAC is considered likely to occur. The remaining 9 systems and their components were screened systematically against the criteria in the methodology; those deemed by the RP to have a significant overall likelihood of failure due to FAC (based on chromium content, environmental parameters and flow conditions) were taken forward to where options to eliminate, reduce or mitigate the FAC risk are considered (Ref. 141).

650. Data from Chinese CPR1000 PWR plants is presented by the RP to support the assigned FAC risk level due to environmental conditions in the AAD [SSFS] and the ABP [LPFHS] (Ref. 140). For these systems dissolved oxygen concentration data is presented, although this is limited to data from a single CPR1000 unit and only average values are presented. No other plant data from CPR1000 or other plants is used to support the evaluation of FAC risk from environmental conditions in any of the systems. The use of additional operational plant data and/or other OPEX would have strengthened the justification for the assigned FAC risk levels based on environmental

conditions for these, and other components, and would add important supporting evidence to the RP's case. I consider this to be part of AF-UKHPR1000-0011 on the use of OPEX in the safety case.

651. Overall, when considering the expectations set out in SAP ECH.1 and NS-TAST-GD-088, I consider that the RP has devised and implemented a suitable methodology to systematically review the degradation threat to the plant from FAC, based on the materials of construction and operating environment, to an extent that is adequate for GDA (Ref. 2, Ref. 6). I judge that the RP has considered an appropriate range of RGP and has presented appropriately underpinned criteria on which to rank the likelihood of failure of SSCs due to FAC for the generic design.

Consideration of Options to Reduce Flow Accelerated Corrosion Risk

652. RO-UKHPR1000-0034 required the RP to identify and document the options considered to eliminate, reduce or mitigate the risks of materials degradation due to FAC in the susceptible areas identified. The RP has set out the options considered for the SSCs it deemed to have a significant overall likelihood of failure due to FAC in 'ALARP Report for FAC Risks in UK HPR1000 Secondary Circuit' (Ref. 141).

653. The RP firstly considers general options for reducing FAC risk, such as increasing the secondary circuit pH to above 10 and increasing the dissolved oxygen concentration. These options are discounted on the basis that the optimum chemistry has been determined holistically to achieve an approach that considers all relevant risks, rather than solely targeting FAC risk reduction. Later in the document, further risk reduction options are considered on a component by component basis. A steel with a higher chromium content ($1.0 \leq Cr \leq 1.5\%$ rather than $0.2 < Cr \leq 0.4\%$) is selected for the Feedwater Heater Drain Recovery System (ACO [FHDRS]) and several components of the ADG [FDTGSS], as well as an increase in the radius to diameter ratio of elbows in the APG [SGBS]. Decisions on other risk reduction options, in particular material changes that would result in significant numbers of dissimilar metal welds, are deferred until detailed layout information is available for UK HPR1000, as the RP argues that conventional safety risks cannot be assessed (and therefore a position cannot be reached on whether the proposed material changes reduce risks SFAIRP) until layout information for these systems is available. Whilst I judge that this is an acceptable position for GDA, I consider that further evidence is required in future to demonstrate that all reasonably practicable modifications to the design and materials selection have been made in order to reduce FAC risk SFAIRP in UK HPR1000, and to demonstrate that reliance is not placed on active chemistry controls where it is reasonably practicable not to do so. I consider this to be an Assessment Finding.

AF-UKHPR1000-0161 – The licensee shall, as part of detailed design, demonstrate that all reasonably practicable improvements to the design and materials selection have been implemented to reduce the risk of flow accelerated corrosion so far as is reasonably practicable. This should include implementing the commitments identified during Generic Design Assessment, and minimising the reliance placed on chemistry mitigations, where it is reasonably practicable to do so.

654. Whilst it is clear that there remains further work to be done to consider and implement possible options to reduce FAC risk as the relevant detailed design information becomes available, I judge that the RP has described and considered a suitable range of available options for reducing FAC risk for the generic design of UK HPR1000. Additionally, I consider that the RP has developed an appropriate FAC risk analysis methodology for the licensee to implement.

Demonstration that Flow Accelerated Corrosion Risks Have Been Reduced SFAIRP

655. I consider that the RP has described and evaluated a suitable range of options for reducing FAC risk in UK HPR1000, based on the information that is available to it at the current design stage. Of the options considered, the RP has identified a number of potential design changes (to the ACO [FHDRS], ADG [FDTGSS] and APG [SGBS]) that it considers are reasonably practicable to implement based on the information available on the generic design, and a range of further measures that require additional consideration once detailed layout information is available. However, the RP has not incorporated the identified changes to the ACO [FHDRS], ADG [FDTGSS] and APG [SGBS] into the Design Reference for GDA, arguing that the secondary circuit systems in question do not form part of the defined scope for GDA, and instead capturing this work as a commitment for the post-GDA stage. I am content with this approach for GDA given that the RP has provided an appropriate methodology, and further work will be required at a post-GDA stage to incorporate the identified changes and develop the FAC analysis; this forms part of AF-UKHPR1000-0161. Despite this Assessment Finding, I am content that an appropriate demonstration that risks have been reduced SFAIRP has been made for GDA and consider that sufficient information has been provided in the extant generic safety case for the licensee to build upon.

656. The RP notes that all components in the secondary circuit of UK HPR1000 are accessible and are designed for repair and replacement, and that for components where the risk of failure due to FAC remains significant, a detailed and regular inspection and maintenance strategy should be developed by the licensee. The definition of a suitable monitoring/surveillance programme is an important consideration for FAC mitigation at any operating reactor, however, it is not expected to be developed during GDA due to the requirement for the availability of detailed design and operating chemistry information, amongst other things. An appropriate programme will need to be defined once this information is available and a subsequent more detailed FAC risk analysis has been carried out. I consider this to be part of the normal business activities expected to be completed at a post-GDA stage.

657. I have considered the expectations set out in SAP ECH.1 and NS-TAST-GD-005, and I judge that the RP has provided suitable and sufficient evidence to support its claim that FAC risks have been reduced SFAIRP for the generic design of UK HPR1000 (Ref. 2, Ref. 6).

4.3.4.3 Strengths

658. The RP has considered an appropriate range of RGP and has presented appropriately underpinned criteria on which to rank the likelihood of failure of SSCs due to FAC for the generic design. Overall, I consider that the RP has developed and recorded an appropriate FAC risk analysis methodology for the licensee to implement and build upon.

659. Additionally, the RP has appropriately considered the main chemistry related threats to the SG tube integrity. Many of the historically encountered problems in this area should be eliminated or reduced in UK HPR1000.

4.3.4.4 Outcomes

660. Based upon the assessment of secondary circuit material integrity and corrosion in UK HPR1000 described in sub-section 4.3.4 above, I have identified one Assessment Finding, which needs to be addressed by the licensee, concerning the demonstration that all reasonably practicable improvements to the design and materials selection have been implemented to reduce the risk of FAC SFAIRP (AF-UKHPR1000-0161). I have also identified an area that is relevant to the Assessment Finding concerning

inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011) raised earlier in this report.

661. I also identified a minor shortfall as discussed in sub-section 4.3.4.2 above.

4.3.4.5 Conclusion

662. Based on the outcome of my assessment of secondary circuit material integrity and corrosion in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

663. I have identified one Assessment Finding as a result of this part of my assessment. Despite this Assessment Finding, I am content that an appropriate ALARP demonstration has been made for GDA and consider that the RP has developed an appropriate FAC risk analysis methodology for the licensee to build upon.

664. The principal SAPs relevant to my assessment of the materials integrity are ECH.1 to ECH.3, EAD.1, EMC.13 and EMC.16. I am content that, overall, the expectations set out in these assessment principles have been met by this part of the generic safety case.

4.3.5 Secondary Circuit Sampling

665. A significant step towards adequate chemistry control is to ensure that representative and reliable sampling and analysis is carried out at an appropriate frequency. ONR’s expectations for chemistry monitoring, sampling and analysis are set out in SAP ECH.4 and NS-TAST-GD-089, and note that adequate provisions should be in place to ensure that the required level of chemistry control for safety will be effectively implemented, including obtaining and maintaining the required quantity and quality of data to support decision making, throughout all phases of the operational life of a nuclear facility (Ref. 2, Ref. 6).

666. In the secondary circuit, achieving representative sampling for some species can be a challenge. Specifically, feedwater oxygen, hydrazine and CPs samples can be particularly affected by the conditions within the sampling lines, which tend to have a significantly larger surface to volume ratio than the pipework from which the sample is drawn. Sample lines may also be relatively long, with sample coolers situated far away from the sample point. Sample compositions can change in the sample line as a result of precipitation or reaction with the sample line surface, resulting in a less representative sample reaching the monitoring equipment. The design of the sampling lines and the careful selection of sampling points is therefore important in achieving representative sampling and analysis.

667. The RP makes the following sub-claim in PCSR Chapter 21 of relevance to secondary circuit sampling; “3.3.10.SC21.10: The chemistry and radiochemistry parameters linked to integrity, reactivity control and radioactivity are sampled and monitored in accordance with the sampling philosophy” (Ref. 5). The sub-claim is supported by a number of arguments and sub-arguments.

668. Similarly to primary circuit sampling, my assessment of the UK HPR1000 secondary side sampling systems has been informed by a TSC (Ref. 34).

4.3.5.1 Secondary Circuit Sampling Systems

669. Sampling and monitoring of the UK HPR1000 secondary circuit steam and feedwater systems is performed by the SIT [FCSS], except for the APG [SGBS], which is sampled via the REN [NSS]. Together these systems deliver representative samples of

fluids from secondary systems which are used to detect impurity ingress and provide information on deviations in plant performance.

Feedwater Sampling

670. The SIT [FCSS] is described in PCSR Chapter 21 and in a dedicated DSR (Ref. 5, Ref. 143). The SIT [FCSS] enables continuous sampling of the secondary circuit steam and feedwater. In addition, grab samples can be taken at a number of locations within the turbine hall for laboratory analysis.

671. The SIT [FCSS] online analysers are located in a central sampling room and are used to continuously monitor cation conductivity, direct conductivity, sodium, oxygen, pH (25 ºC) and hydrazine. Samples are first cooled in the turbine hall at the point of take-off before being directed to the sample room for analysis by online instruments. Manual samples are taken locally near to the sampling points using a portable cooler, and samples from the condenser trays are continuously pumped through analysers mounted on local racks in the turbine hall. After being analysed by online instruments these sample flows cannot be returned to the secondary systems due to possible contamination by chemical reagents or other impurities. Therefore, sample water is discharged into the CI Waste Fluid Collection System [SEK].

672. Online analyser data is sent to the Main Control Room and is used to control automatic dosing of the secondary circuit. The DSR lists the following sampling point locations for the online analysers (Ref. 143):

- Condensate Extraction Pump Discharge, CEX [CES]
- Condensate Extraction Pump Discharge (after ATE [CPS], CEX [CES]
- LP Heaters Drains Pump Discharge, ACO [FHDRS]
- Deaerator Recirculation Pump Discharge, ADG [FDTGSS]
- Deaerator Downcomers, ADG [FDTGSS]
- HP Heaters Drains, AHP [HPFHS]
- MSR Drain Pump Discharge, MSR system
- Main Feedwater Line, ARE [MFFCS]
- MSLs from Each SG, Main Steam and Drainage System for Conventional Island
- Condenser Trays, CEX [CES]

673. Six of these locations are constantly monitored with other areas only being used occasionally to help operators in detecting the source of impurity ingress. The six locations are the Condensate Extraction Pump Discharge, Deaerator Downcomers, Deaerator Recirculation Pump Discharge, Main Feedwater Line, Condenser trays and MSLs from each SG.

674. Regarding the representativeness of samples, the system is designed so that continuous flow can be maintained in the sample lines in order to obtain a homogenous sample. To ensure homogeneity, it is important to maintain a continuous turbulent sample flow by achieving a Reynolds number (Re) of greater than 4000. Sampling flow and pipe diameters have been selected to ensure turbulent flow is maintained (Reynolds number Re >4000). The layout of the pipework minimises low points, bends and dead ends in order to avoid the deposit of impurities, and sampling lines are made from 316 stainless steel. Hydrazine will react with oxygen during sample transport and the rate of reaction will increase with higher temperatures. This means it is difficult to obtain a representative concentration of hydrazine or oxygen in the secondary circuit. To minimise this effect, the RP claims that the sampling lines between the feedwater sampling point and primary sample cooler are as short as possible. To obtain representative sampling of suspended solids and ionic solids, a CPs sampler has been implemented. System transients may be missed when taking a grab sample, so by taking an integrated sample over time a better understanding of the

transport of CPs to the SGs is obtained. The CPs sampler importantly has a flow totaliser which will accurately determine the total volume of sample that has passed through the sampler. My TSC reviewed the design considerations made by the RP to ensure that representative sampling is achieved; I considered these to be appropriate (Ref. 144).

675. Cation conductivity is continually monitored from the Condenser Trays in the CEX [CES] through analysers mounted on a local rack. This measurement is used to detect condenser leaks with an alarm being raised in the main control room if the conductivity exceeds a set limit. Cation conductivity relies on a cation exchange column which can become exhausted, thereby requiring regular replacement of the cation column. Although the system appears adequate for the purposes of routine chemistry sampling and monitoring, my TSC noted that further evidence to support its adequacy in providing timely detection of condenser leaks is required, for example consideration of direct conductivity instead of cation conductivity. Additionally, I judge that consideration should be given, at the site-specific stage, to whether automatic actions are required on detection of impurity ingress from a condenser leak (for example, a system of the type which trips the reactor if a high level of impurity ingress is detected in the boiler feedwater). I consider this to be an Assessment Finding.

> AF-UKHPR1000-0162 – The licensee shall justify the capability of UK HPR1000 to provide timely detection of condenser leaks, and shall review the requirements for automatic actions in the case of leaks that could impact nuclear safety.

676. Manual samples are taken from the following locations:

- Bled steam to LP Heaters, ABP [LPFHS]
- Bled steam to HP Heaters, AHP [HPFHS]
- Makeup Line to Condenser, CEX [CES]
- Each end of Feedwater Deaerator Tank and Bled Steam to Deaerator, ADG [FDTGSS]
- Drain Receivers of MSRs, MSR System
- Steam Pipes to LP Cylinder, Turbine Steam and Drain System (TSDS)

677. As well as the sampling locations described above, manual grab samples can be taken from any of the sampling locations used for continuous online analysis at the analyser panel in the sample room. To ensure grab samples are cooled to a safe temperature and pressure for operators, a portable cooler is used. The cooler has gauges to indicate pressure, temperature and cooling water flow.

678. Due to saturated steam being sampled as a two phase fluid, the RP notes that isokinetic sampling will be employed. This is standard industry practice for sampling of MSLs. It is also common to use isokinetic sampling for particulate samples; however, my TSC noted in their assessment that it is not clear from the safety case if the CPs samplers will use isokinetic probes or not. I consider that further justification for the adequacy of the SIT [FCSS] in providing representative sampling of CPs would strengthen the safety case. I therefore consider this to be a minor shortfall.

679. Overall, the RP appears to have made appropriate considerations to ensure representative sampling is achieved in the SIT [FCSS]. ASTM standards have been followed, although, as discussed for the REN [NSS] in sub-section 4.2.8.1, these standards are not specific to nuclear systems. For the SIT [FCSS] however, the RP has included comparisons of its system to other established secondary circuit sampling system designs and RGP, including UK EPR™ and UK AP1000®, in its safety case documentation which helps to justify design decisions.

Steam Generator Blowdown System Sampling

680. The part of the REN [NSS] which enables sampling and monitoring of the APG [SGBS] is described in the REN [NSS] DSR (Ref. 116). For the three APG [SGBS] blowdown sampling lines, a sodium meter (common, one meter for three APG [SGBS] sampling lines), three pH meters (one for each line) and three conductivity meters (one for each line) are provided for online monitoring of the blowdown water quality. Additionally, radioactivity is continuously monitored on each of the three lines via the PRMS to monitor any primary-to-secondary transfer of radioactivity, most likely due to SG tube leakage or rupture. Secondary fluid is extracted from the APG [SGBS] sampling lines and routed via gamma activity measurement equipment such that each SG is monitored separately and continuously for radioactivity carryover to the secondary side.

681. The APG [SGBS] has four sampling lines coming from the two filter demineraliser trains (one after each of the cation beds and one following each of the resin trap filters). In order to monitor the APG [SGBS] resin saturation, a sodium meter (common, one meter for two cation exchanger sampling lines) and four conductivity meters (two for direct conductivity measurement and two for cation conductivity measurement) are provided for online monitoring. Grab samples can also be taken from branch sampling lines if online analysers are unavailable. The sampling water taken by the REN [NSS] is returned downstream of the blowdown control valve for treatment and recycling.

682. Overall, the RP’s choice of chemical parameters and sample locations appear adequate, with the reasoning for these choices suitably described in the DSR. However, just as for the rest of the REN [NSS] (described in sub-section 4.2.8.1), no OPEX was presented in the generic safety case to support design decisions and the justification for the representativeness of samples. I judge that the safety case would be improved if plant data or other OPEX (from CPR1000 or other PWR designs) was presented to strengthen the justification that the REN [NSS] can provide representative sampling for all chemistry and radiochemistry parameters; I consider this to be part of AF-UKHPR1000-0011 on OPEX in the safety case.

4.3.5.2 Strengths

683. Overall, I note that a number of beneficial features have been included in the design of the secondary circuit sampling systems, including:

- For the SIT [FCSS], temperature conditioning in close proximity to the sample point has been implemented to ensure consistent and accurate sampling of oxygen and hydrazine.
- Isokinetic sampling of the MSLs will be used which follows industry good practice.
- To obtain accurate representative samples, sampling lines have been designed to maintain turbulent flow (Reynolds number Re >4000).
- The use of online analysis for the majority of parameters in order to detect sudden transients.

4.3.5.3 Outcomes

684. Based upon the assessment of secondary circuit sampling and monitoring in UK HPR1000 described in sub-section 4.3.5 above, I have identified one Assessment Finding, which needs to be addressed by the licensee, concerning the capability of UK HPR1000 to provide timely detection of condenser leaks (AF-UKHPR1000-0162). I have also identified an area that is relevant to the Assessment Finding concerning inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011) raised earlier in this report. I also identified a minor shortfall in this part of my assessment.

4.3.5.4 Conclusion

685. Based on the outcome of my assessment of secondary circuit sampling and monitoring in UK HPR1000, and in line with the expectations set out in SAP ECH.4, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA. For the majority of secondary circuit chemical parameters, adequate evidence has been presented to support the RP's claims that representative sampling can be achieved.

686. I have identified an Assessment Finding as a result of this part of my assessment.

4.3.6 Start-up and Shutdown Chemistry

687. During any shutdown the secondary circuit will be taken from normal operating conditions of high temperature and pressure to almost ambient conditions. As the secondary circuit is nominally non-radioactive, there is no rapid increase in the concentrations of soluble and particulate radionuclides, although a significant increase in iron in the coolant is expected and, as such, the concern is more with maintaining adequate chemistry control during the transient. Similarly, start-up periods can pose such challenges and the main difficulty is with establishing and maintaining the correct chemistry.

688. The RP's safety case structure has a report for chemistry at-power, and a separate report which sets out details of the start-up and shutdown chemistry proposed for UK HPR1000 (Ref. 28, Ref. 29). Beneath these two documents are the lower level reports which focus on individual topics, such as the choice of secondary circuit pH raising agent, which provide evidence to support both the at-power and start-up and shutdown reports.

689. The RP does not make separate claims and arguments on secondary circuit start-up and shutdown chemistry, however many of the secondary circuit claims and arguments, as described in sub-section 4.3.1, apply during all modes of operation, including in start-up and shutdown periods.

4.3.6.1 Control of Chemistry During Start-up and Shutdown

690. The 'Topic Report on Start-up and Shutdown Chemistry' notes that the main objective of feedwater pH control during start-up and shutdown is to minimise the general corrosion of secondary circuit materials, minimise transport of CPs into the SGs and prevent the formation of sludge and localised corrosive environments in the SG (Ref. 29). This is in order to minimise the corrosion risk to the SG tubes and internals during subsequent power operation.

691. During reactor start-up, impurities within the secondary circuit are often present in much higher concentrations than during normal operations. The concentration of impurities needs to be controlled to minimise the inventory of CPs in the feedwater which could be transported to the SG. This is achieved by operating the ATE [CPS] and the APG [SGBS] during start-up (these systems are described in sub-section 4.3.2), and minimising impurity ingress from the make-up water and chemicals through relevant specifications. Even though control of impurity levels is important to ensure optimum start-up conditions, the risks from impurity ingress are less during shutdown compared to normal operation, due to the lower temperatures. Hence the RP argues that only diagnostic parameters are needed for sodium, cation conductivity, chloride, fluoride, sulphate, silica and suspended solids during these periods (although oxygen, cation conductivity and sodium are monitored online, in the same way as they are in power operations). The RP notes that if unexpected very high impurity ingress occurs (chloride, sulphate, sodium), and if concentrations and/or cation conductivity values in

the SGs are above the highest Action Level values of power operation, the start-up process will be stopped and the SGs drained and cleaned. I am content with the arguments set out.

692. Hydrazine dosing and removal of bulk oxygen by the ADG [FDTGSS] is performed during start-up and shutdown to minimise CPs formation and transport. The RP notes in response to RQ-UKHPR1000-0718, that on the basis of OPEX from its operating PWR units, the feedwater oxygen content will be monitored and limited at < [redacted] mg kg^{-1} during start-up when the temperature is above 120 °C, but that it is not required to be a control parameter below this temperature (Ref. 3). This is in line with VGB guidelines and practices at other operating PWRs.

693. The RP notes that during start-up and shutdown the SG steam is by-passed to the condenser through the TBS [GCT], and two-phase flow therefore does not occur. The RP claims that this therefore negates the need for control parameters relating to secondary circuit pH. Considering the effect of pH on iron transport, and its impact on subsequent power operation, however, the feedwater $pH_{25\,°C}$ operating window is specified to be the same as during power operation ([redacted]). Feedwater pH control during start-up and shutdown is achieved with ammonia dosing from the SIR [CRIS] in the same manner as during power operations. Whilst the operating window appears reasonable, given the effect of pH on iron transport and considering practices at some operating PWRs, the licensee should consider whether a control parameter on pH is needed. I consider this to be a minor shortfall as this is only an operational change.

694. Whilst the start-up and shutdown chemistry controls for the secondary circuit are described in the safety case at a relatively high level, with only some of the parameter values specified, I am content that the general approach set out is reasonable and provides a suitable basis on which the licensee can develop detailed procedures during the site-specific stages.

4.3.6.2 Steam Generator Storage and Lay-up

695. Once cooled and stabilised in Normal Shutdown with RIS [SIS] in NS/RIS-RHR mode, the secondary circuit needs to be placed into a suitable storage condition which, with the exception of the SGs, means a dry lay-up. For the SGs two types of lay-up are proposed by the RP, wet lay-up and dry lay-up, depending on the duration of the shutdown and the access requirements. In PCSR Chapter 21 and the ‘Topic Report on Start-up and Shutdown’, the RP provides an overview of the aims and processes for SG lay-up during periods of reactor shutdown (Ref. 5, Ref. 29). During wet lay-up the main objectives are stated to be:

- Maintaining alkaline reducing conditions in the SGs to prevent corrosion risk induced by oxygen ingress.
- Promoting the return of impurities from SG deposits and crevices.

696. The topic report notes that for a short shutdown of less than a week, generally the SGs do not need to be drained, and the water quality required is the same as during normal power operation (as assessed in sub-section 4.3.3). For a lay-up of more than one week, additional ammonia and hydrazine are injected into the SGs to limit the risks of corrosion. The RP notes that wet lay-up is generally preferred to dry lay-up, because during dry lay-up it is difficult to ensure that the SG internal surface is completely dried through air sweeping.

697. For wet lay-up of more than one week, the secondary circuit is under maintenance shutdown mode with a temperature of below 60 °C and is open to the air. In this condition there is a risk of general corrosion of the carbon steel secondary circuit materials. The RP notes that the chemistry parameters associated with the SG lay-up include pH, ammonia, hydrazine, and impurities (sodium, chloride and sulphate). I

raised RQ-UKHPR1000-0718 to clarify which parameters would be controlled and to what limits (Ref. 3). The RP's response noted that there will be no control parameters associated with secondary circuit chemistry control during shutdown in UK HPR1000, only diagnostic parameters.

698. The RP argues that the low temperature and heat-flux during shutdown will not lead to concentration and hideout of impurities, and therefore the localised corrosion risks to the SG tubes and internals can be considered minimal. Therefore sodium, chloride and sulphate are monitored primarily to look for increased levels that may have an impact on subsequent power operation. The RP also notes that it considers there is low risk to the integrity of the SGs due to general corrosion within the short duration of a lay-up phase, therefore pH, ammonia and hydrazine are monitored but are not considered as control parameters. Whilst these arguments appear reasonable, no expected values for the diagnostic parameters were provided. The licensee will need to specify and further justify the wet lay-up chemistry specifications for UK HPR1000 as part of its normal business safety case development activities.

699. Dry lay-up can be implemented when maintenance activities require it, as well as before the first fill of the SGs. I raised RQ-UKHPR1000-0563 to clarify the environmental conditions that need to be maintained during dry lay-up, as condensation can be detrimental due to pollution from airborne contaminants, which can lead to the generation of very high local impurity levels (Ref. 3). The RP responded that relative humidity is controlled to limit the condensation of impurities, especially chloride, which could be present in the air. The specific relative humidity limit for UK HPR1000 will be provided in the site phase, but the RP notes that for FCG3, it is limited below [redacted]% (at 20 °C) based on the operating experience of CPR1000 units. Whilst I am content with the response and the case made for GDA (as this is largely an operational matter), the licensee will need to specify and justify dry lay-up chemistry specifications as part of its normal business safety case development activities. This should include all stages of installation, commissioning and operations.

4.3.6.3 Strengths

700. The RP has started to identify and consider the chemistry risks and requirements for start-up and shutdown in the secondary circuit.

4.3.6.4 Outcomes

701. Based upon the assessment of secondary circuit start-up and shutdown chemistry in UK HPR1000 described in sub-section 4.3.6 above, I have not identified any Assessment Findings. I have identified one minor shortfall.

4.3.6.5 Conclusions

702. Based on the outcome of my assessment of the secondary circuit start-up and shutdown chemistry in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA, which can be built upon by the licensee.

703. As part of my assessment, I referred to the expectations set out in the SAPs ECH.1 and ECH.3, which note that safety cases should consider all chemistry effects and expect a demonstration that chemistry is controlled adequately (Ref. 2). I also referred to NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the generic safety case.

4.4 Auxiliary Systems

4.4.1 Spent Fuel Pool

704. Spent fuel is discharged from the reactor and transferred to a SFP where decay heat is removed, and the used fuel is stored for a period. For UK HPR1000, spent fuel assemblies are stored in the SFP for a number of years pending transfer to dry cask storage in an on-site Spent Fuel Interim Storage (SFIS) facility. PCSR Chapter 28 describes the UK HPR1000 Fuel Handling and Storage System (PMC [FHSS]) and associated activities, and PCSR Chapter 29 provides information on SFIS (Ref. 27, Ref. 145).

705. My assessment of the UK HPR1000 SFP and associated systems focussed primarily on the chemistry control of the systems and how provisions have been made in the design to accommodate these requirements. I considered the following systems and associated activities to be relevant to the scope of my Chemistry assessment:

- SFP and PTR [FPCTS]
- Fuel Transfer Compartment
- Reactor Pool

4.4.1.1 Overview

706. The SFP is a large, stainless steel-lined pool of borated water containing underwater storage racks which hold the irradiated fuel assemblies that have been discharged from the reactor. New fuel and any damaged fuel are also stored in the SFP. The storage racks are designed to maintain the stored fuel in a sub-critical geometry and have affixed solid neutron absorbers to provide a sub-critical margin. The SFP is sized to store the spent fuel assemblies discharged from ten refuelling cycles and a full-core emergency offloading. Five cells are provided with failed fuel assembly storage filters to store damaged fuel assemblies.

707. Figure 6 shows the general layout of the UK HPR1000 PMC [FHSS]. The SFP can be connected to the Fuel Transfer Compartment via a sealed gate. The Fuel Transfer Compartment incorporates the Fuel Transfer Facility which allows fuel assemblies to be passed to and from the BRX during refuelling operations. As the SFP mixes with the primary coolant during refuelling activities, the quality of SFP water is controlled to the same level as the primary coolant, including the soluble boron concentration and impurity limits.

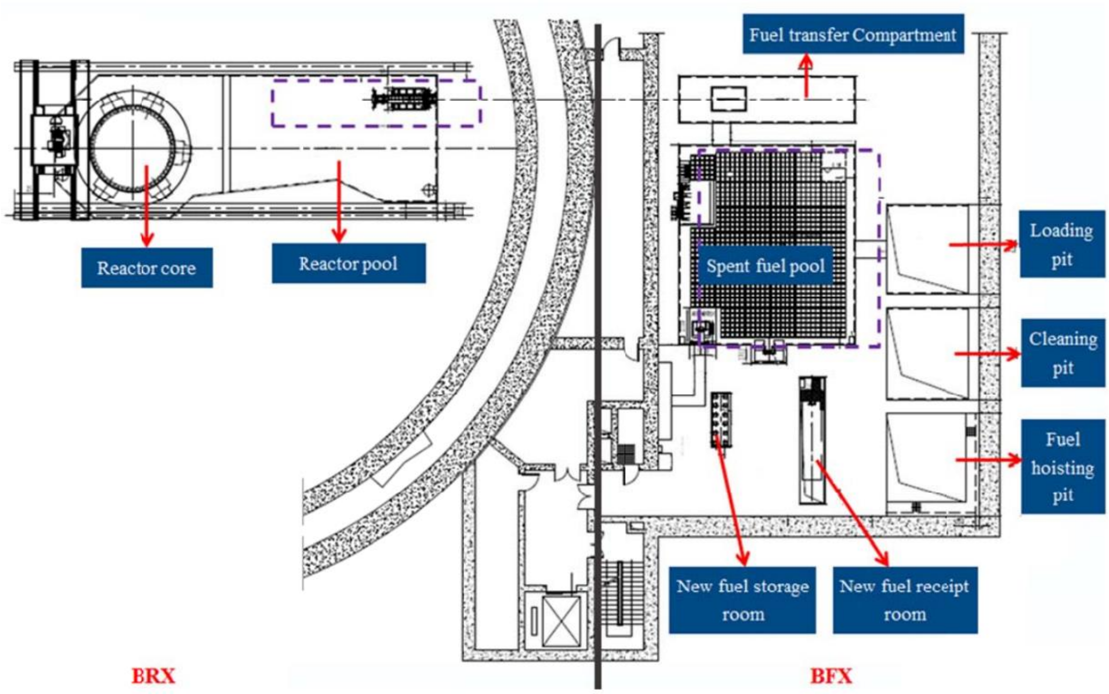

Figure 6: General overview of the Fuel Handling and Storage System **(Ref. 27)**

708. The RP makes the following sub-claims relating to SFP chemistry in PCSR Chapter 21 (Ref. 5):

- 3.3.10.SC21.6: Reactivity is controlled in the SFP through material selection.
- 3.3.10.SC21.7: Radioactivity level in the auxiliary systems is reduced SFAIRP to minimise worker and public dose.

709. The sub-claims are supported by a number of arguments and sub-arguments, including that the corrosion of auxiliary system materials is minimised SFAIRP through material selection and chemistry control, and that the radioactivity level in the SFP is controlled by the PTR [FPCTS] during normal operation to minimise the accumulation of radionuclides.

4.4.1.2 Purification

710. The PTR [FPCTS] is designed to remove impurities from the SFP to reduce corrosion of the spent fuel assemblies and the structural components through filtration, and to maintain visibility in the SFP through surface skimming. The system can also perform purification of other pools and the IRWST. Aside from discharge and make-up, the PTR [FPCTS] is the only means to maintain chemistry control and to minimise radioactivity in the SFP. The purification unit is located off the discharge line of the SFP purification pump and consists of a filter located upstream of the demineraliser to remove suspended solids, a mixed bed resin demineraliser to remove anion and cation impurities and a resin trap filter to prevent any resin fines entering the SFP.

711. The system is also required to remove decay heat from the irradiated fuel in the SFP in order to maintain the temperature in the pool within an appropriate range; the RP specifies a temperature range of 15 °C to 50 °C during normal operations in order to minimise evaporative loses. The PTR [FPCTS] cooling subsystem has three cooling loops situated in physically separated areas of the fuel building. Each cooling loop has one cooling pump and one heat exchanger which is supplied with cooling water from either the RRI [CCWS] or the Extra Cooling System (ECS), as shown in Figure 7.

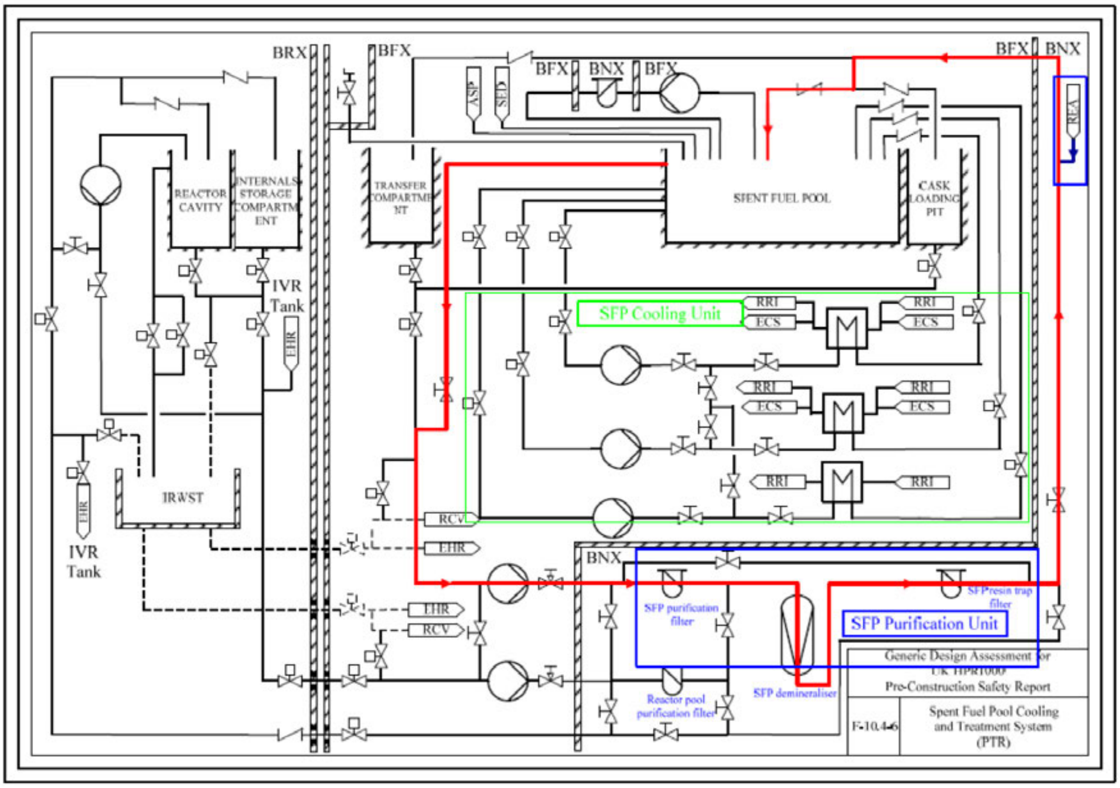

Figure 7: Simplified flow diagram of the PTR [FPCTS] **(Ref. 146)**

712. In addition to purification and cooling of the SFP, the PTR [FPCTS] is also the route to add make-up water from the NI DWDS and borated water from the REA [RBWMS] to the pools to compensate for evaporative loses. The system also facilitates sampling of the SFP for chemical or radiochemical analysis via the REN [NSS].

713. When the reactor pool purification pump and filter are not in service, they can be used as a backup for the SFP purification pump and filter. The purification loop is connected to the TES [SWTS] for the removal of spent resins produced by the demineraliser. In RQ-UKHPR1000-0495, I asked about any provision for purification of the SFP during resin replacement (Ref. 3). The RP's response noted that when the resin needs to be replaced, the purification unit is bypassed and the concentration of radionuclides in the SFP would increase to the upper limit after approximately 68 hours of the purification unit being out of use (calculated with conservative assumptions). The RP argues that resin replacement can be performed within this timescale, and that therefore no alternative purification provision is required. I consider this argument to be reasonable.

714. The RP provided a detailed DSR containing calculations for the sizing of the PTR [FPCTS] filters and demineraliser, supported by operating data from CPR1000 plants which operate with a similar PTR [FPCTS] design (Ref. 146). Whilst the calculations appear robust and conservative, the plant data presented to support the purification ability of the system was limited in scope. Whilst I am content with the evidence presented for GDA, further OPEX or other supporting information would strengthen the justification for the purification ability of the PTR [FPCTS]; I consider this to be part of AF-UKHPR1000-0011 on use of OPEX in the safety case. The response to RQ-UKHPR1000-0840 clarified that the sizing of the system assumes the presence of fuel assemblies containing pinhole cladding failures in all five of the failed fuel storage cells in the SFP (Ref. 3).

715. I note that a cross-cutting RO (RO-UKHPR1000-0056) was raised during Step 4 of GDA regarding the safety case for the handling of spent fuel (Ref. 4). As part of its response to the RO, the RP modified the design of the BFX to enable the use of a

gantry crane for spent fuel movements. The modification results in a decrease in the storage capacity of the SFP, from 1020 assembly storage cells to 975 cells, and an increase in the SFP water volume of approximately 4 m^3 (around 0.3% of the water volume prior to the design modification). The RP claims that the capacity of the PTR [FPCTS] remains sufficient to treat the SFP water since the storage capacity of the SFP is decreased and therefore fewer impurities are introduced into the pool. Additionally, the increase in water volume to be treated is negligible. I am content with these arguments, although I note that the DSR will not be updated to reflect the effects of the design modification on the PTR [FPCTS] until the site-specific stage (Ref. 146).

716. Overall, the design of the PTR [FPCTS] is functionally very similar to those currently in use at other PWRs and I consider that the RP has provided a coherent, adequate safety case to justify the chemistry functions of the system.

4.4.1.3 Operational Chemistry

717. The control of SFP chemistry in UK HPR1000 is a key factor in preventing cladding corrosion of new and used fuel and to minimise the spread of radioactivity arising from failed fuel in the SFP. The RP propose the following chemistry control parameters for the SFP during normal operations:

Table 14: Spent Fuel Pool Chemistry Limits During Power Operation **(Ref. 30)**

Control Parameter	Units	Operating Window
Total B	mg kg^{-1}	1300~1400
Enrichment of ^{10}B	at%	35~39
Chloride	mg kg^{-1}	0.15
Fluoride	mg kg^{-1}	0.15

718. Chloride and fluoride are monitored and controlled to reduce corrosion of the spent fuel assemblies and the structural components through filtration; according to Auxiliary Water Chemistry Specifications grab sampling is performed twice per month or once per week during refuelling operations (Ref. 147). Other impurities, including sulphate, sodium and total silica, are also monitored as diagnostic parameters. The SFP chemistry will be disturbed either due to water transfers or mixing with other water sources, especially during refuelling, or by activity released into the pool water from stored spent fuel. As noted in sub-section 4.4.1.2, I consider that a suitable justification for the capability of the PTR [FPCTS] to maintain the chemistry and deal with such disturbances has been provided by the RP.

719. Total boron concentration and ^{10}B enrichment in the SFP are control parameters; grab sampling is performed once per week or twice per week during refuelling operations (Ref. 147). Whilst the soluble boron in the SFP is not claimed to be required to maintain sub-criticality for fuel stored in the racks (due to the presence of fixed neutron absorbers), it provides an extra margin to criticality and during refuelling operations the SFP water comes into contact with the primary coolant, so the soluble boron must be of sufficient concentration to prevent dilution. The boron concentration in the SFP is maintained through addition of borated water from the REA [RBWMS]. Boronated make-up water from the IRWST is also transferred to the reactor pool to prevent dilution of the primary coolant when the reactor vessel is opened.

720. The control of criticality in the SFP is achieved with the use of fixed neutron absorbers. Neutron absorbers are fixed to the outside of cells in the SFP storage racks by thin

stainless steel plates and enable increased storage capacity whilst maintaining criticality safety margins. Neutron absorber materials based on polymers used in the past have shown significant degradation when exposed to gamma radiation in SFP environments, releasing large quantities of silica to the SFP water or experiencing corrosion damage that threatened the neutron attenuation properties of the material.

721. The RP produced 'Material Selection Report of Neutron Absorber of UK HPR1000' (Ref. 148). An aluminium boron carbide (Al-B_4C) Metal Matrix Composite (MMC) is selected as the preferred neutron absorber material, based on a detailed optioneering process and supported by relevant OPEX. The RP presents OPEX to demonstrate good corrosion resistance of MMC neutron absorbers in operating plants since 2003. In addition, details of laboratory corrosion testing carried out on the proposed material are provided, alongside details of in-service monitoring of the corrosion performance of the MMC to be carried out using a coupon tree situated in the SFP. I am content with the response provided and consider that an appropriate justification of the material selected for the fixed neutron absorbers has been made.

4.4.1.4 Control of Radioactivity

722. During refuelling of a reactor, radioactivity is transported around a number of plant systems, including the SFP. The SFP systems will become contaminated with activity either due to mixing with other water sources during refuelling, or from activity gradually released into the SFP water from stored spent fuel. The materials choices and surface finishes can affect the retention of activity in areas where radioactive species have the potential to be accumulated.

723. In PCSR Chapter 21, the RP notes that materials selection in the SFP and related systems aims to minimise corrosion and therefore to minimise radioactive source terms (Ref. 5). I raised RQ-UKHPR1000-0495 during Step 3 of GDA to clarify the RP's justification for SFP materials choices. (Ref. 3) The response noted that stainless steels are selected for the SFP liner, as well as for the PTR [FPCTS] piping and the tubing of the PTR [FPCTS] cooling system heat exchangers (tube side) which is in contact with the SFP water. The RP claims that this provides excellent corrosion resistance under SFP conditions, adding that materials have been selected and relevant degradation risks have been considered in accordance with worldwide PWR practices. I am content with the materials selected and the justification provided from a Chemistry perspective.

724. In addition to bulk materials selection, surface finish can have an important impact on the retention of activity in areas where radioactive species have the potential to be accumulated. I raised RQ-UKHPR1000-1633 to ask what consideration had been given to surface finishes in the SFP to reduce the dose arising from the accumulation of activated CPs (Ref. 3). The SFP and BRX pools' finish is stated to be controlled to ensure surface roughness is no more than 3.2 μm, which the RP claims is the optimum finish to reduce dose rates from the pools SFAIRP and that tighter controls would not be reasonably practicable. I consider this to be an acceptable position for GDA, but judge that further justification would strengthen the safety case as it develops. I consider this to be a minor shortfall.

725. The PTR [FPCTS] also forms a key part of the RP's arguments relating to the control of radioactivity in the SFP. During normal operation, the purification flowrate of the PTR [FPCTS] is approximately 90 $t\ h^{-1}$ and therefore the time required to purify the SFP once is about 15 h. This will rapidly reduce soluble and insoluble radioactivity in the SFP water introduced by mixing with the primary coolant or released from failed fuel. The exception to this is the ^3H concentration within the SFP water which would increase at every refuelling outage, but can be controlled by discharge and make-up within acceptable limits.

726. At shutdown the radioactivity in the primary coolant will mix with the SFP water. The concentration that is present when the RPV is opened is therefore important to control as this directly influences the resultant concentrations in the SFP and reactor pools. The RP has specified limits which must be met before the SFP can be connected to the reactor pool. These are detailed in sub-section 4.2.10.4. I consider these to be adequate to minimise radioactivity transfer to the SFP.

727. I raised RQ-UKHPR1000-0840 to query how the RP intends to manage ^{3}H in the SFP to minimise operator exposure (Ref. 3). The response noted that the gaseous ^{3}H that evaporates from the tritiated water in the SFP is minimised through optimisation of the temperature and the environmental conditions of the SFP. The specified temperature range of 15 °C to 50 °C during normal operations is set in order to minimise evaporative loses. When I queried whether a ^{3}H limit would be set for the SFP, the RP responded that no limit is currently specified for UK HPR1000 but that a target value of [redacted] MBq/t is used in CPR1000 plants. My assessment of ^{3}H control in the primary circuit in sub-section 4.2.3.3 explains that the RP has identified the need for a ^{3}H limit at the end of a cycle prior to the removal of the vessel head, however it has not identified a numerical limit during GDA. I noted in sub-section 4.2.3.3 that I consider there is a need for a ^{3}H limit prior to head-lift, and I similarly consider that a limit or condition may be required for ^{3}H activity in the SFP to limit operator exposure SFAIRP during refuelling. I consider this to be a minor shortfall.

728. Other radiochemistry parameters are also monitored and controlled in the SFP. PCSR Chapter 21 notes that during normal operations, the radioactivity of the SFP water needs to be controlled continuously to ensure that personnel radiation exposure is minimised SFAIRP when handling and lifting the spent fuel. Total gamma and radiochemistry species including ^{133}Xe, ^{131}I, ^{58}Co and ceasium-137 (^{137}Cs) are selected as control parameters on the basis of OPEX from CPR1000 units, however no proposed limits for these parameters have been provided in the generic safety case. Limits should be specified as part of normal business safety case development by the licensee.

4.4.1.5 Strengths

729. The design of the PTR [FPCTS] is functionally very similar to those currently in use at other PWRs and the materials choices are consistent with RGP. I consider that the RP has provided a coherent, adequate safety case to justify the chemistry functions of the system.

4.4.1.6 Outcomes

730. Based upon the assessment of the SFP chemistry in UK HPR1000 described in sub-section 4.4.1 above, I have not identified any Assessment Findings. I did identify an area that is relevant to the Assessment Finding concerning inclusion of robust operational plant data to support safety case claims and arguments (AF-UKHPR1000-0011) raised earlier in this report, however. I have also identified two minor shortfalls.

4.4.1.7 Conclusions

731. Based on the outcome of my assessment of SFP chemistry in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

732. During my assessment of SFP chemistry, I referred to the guidance set out in the SAPs (Ref. 2). The main SAPs of relevance were 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. 6).

4.4.2 Component Cooling Water Systems

4.4.2.1 Overview

733. The UK HPR1000 has a large number of pumps and heat exchangers inside the NI which, together, produce significant quantities of reject heat. In order to assure safe operation and function of these components, a heat removal system is required. In the UK HPR1000, these heat removal functions are provided by four closed water cooling systems.

734. My assessment has focussed on the RRI [CCWS]. This system has the largest number of users including SSCs of the highest nuclear safety significance, such as the primary circuit RCPs and RIS [SIS]. As each system has similar functions, main component materials and operating conditions, I consider it to be representative of all the closed cooling water systems.

735. The RRI [CCWS] provides cooling for users of the NI systems (including nuclear auxiliary systems and safety-classified systems) under normal operating and fault/accident conditions.

736. The closed cooling water systems, with the exception of the ECS are described in Section 6.4 of PCSR Chapter 21 (Ref. 5). Further details of the RRI [CCWS] are contained within the SDM (Ref. 149, Ref. 150, Ref. 151).

737. The RRI [CCWS] cools a number of important components during normal reactor operations and shutdowns, ranging from Class 1 (for example the RIS [SIS]) to non-safety (such as the heat exchangers of the TEP [CSTS]).

738. The RRI [CCWS] consists of three separate and independent trains (A-C) located in separate buildings. The configuration of the system is such that the RRI [CCWS] loops which cool the trains of the users are independent. For example, each train provides cooling to the heat exchangers of the three PTR [FPCTS] trains and the three RIS [SIS] trains. The RRI [CCWS] train A and B are very similar in design; an overview of train A is provided in Figure 8. Train C differs in that it has only one pump, one Essential Service Water System (SEC) cooled heat exchanger, and the only user outside its safeguard building is the PTR [FPCTS].

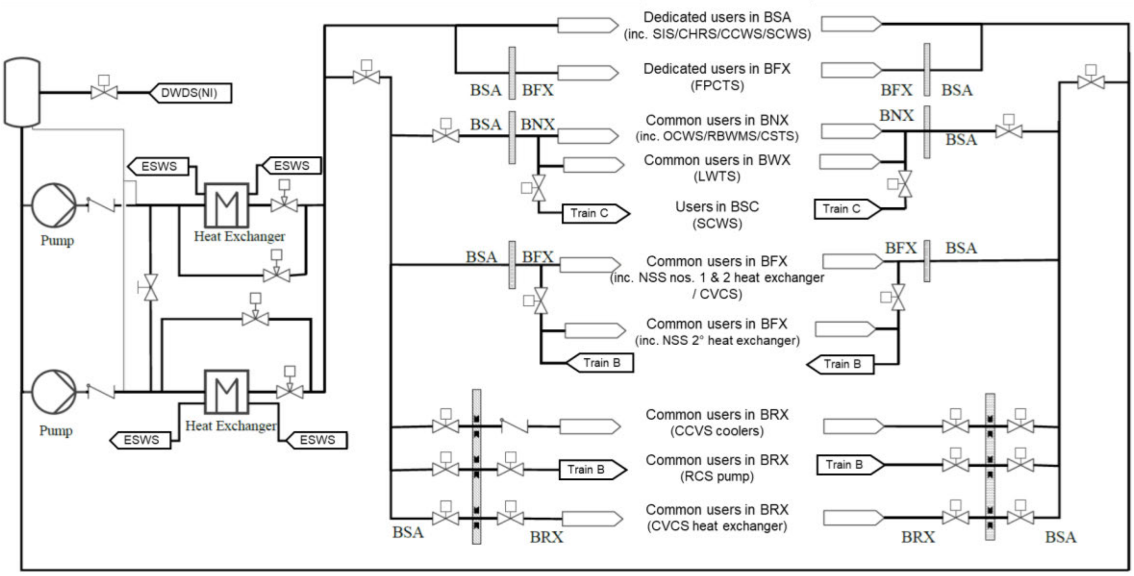

Figure 8: Schematic of the UK HPR1000 RRI [CCWS] Train A, located within Safeguard Building A (BSA), supplying users in BFX, BNX, BRX, Radioactive Waste Processing Building (BWX)

739. Detailed assessment of this system was not started during Step 3 of GDA. During Step 4 of GDA the assessment of the RRI [CCWS] concentrated on the following aspects of the RRI [CCWS] design:

- Chemical conditioning regime, especially for the large range of corrosion mechanisms possible.
- Evidence regarding fouling and scale growth provisions.
- Chemistry control and addition provisions (such as sampling arrangements).
- Leaks into the RRI [CCWS], especially from active sources (including controls, mitigation, remedial actions), and leaks from the RRI [CCWS], especially to sources where there is a risk of boron dilution or contamination with RRI [CCWS] conditioning agents.

740. The RP’s overall case for the UK HPR1000 RRI [CCWS] is presented in a number of documents at various levels within their hierarchy of safety case documentation. Chapter 10 of the PCSR provides details of the system engineering and design, but there are important links to other PCSR chapters, including Chapter 21 (Ref. 21, Ref. 5). Beneath the PCSR there are a number of more detailed documents. The principal documents of relevance to my assessment include:

- ‘Application of Alkaline Agent in the Closed Cooling Water System’ (Ref. 152)
- ‘RRI-Component Cooling Water System Design Manual Chapter 4 System and Component Design’ (Ref. 150)
- ‘RRI-Component Cooling Water System Design Manual Chapter 6 System Operation and Maintenance’ (Ref. 151)
- Radiation and Contamination Monitoring Topic Report (Ref. 153)

741. The principal SAPs relevant to my assessment of the RRI [CCWS] are chemistry (ECH.1, ECH.3, ECH.4) and ageing and degradation (EAD.1 and EAD.2).

4.4.2.2 Operating Chemistry and Materials

742. The RP makes the following sub-arguments in Chapter 21 of the PCSR for the RRI [CCWS] operating chemistry (Ref. 5):

- 3.3.10. SC21.5-A1: The corrosion of auxiliary system materials is minimised SFAIRP through material selection and chemistry control.
- 3.3.10. SC21.5-A1.3: Corrosion inhibitor is added to the closed cooling water systems to minimise material corrosion.

743. I consider these arguments to be reasonable and what I would expect, given the objective for chemistry control in the RRI [CCWS] is predominately to mitigate the risk of materials degradation.

744. ‘Application of Alkaline Agent in the Closed Cooling Water System’ is the RP’s principal submission which contains a justification for the chemistry regime selected for the UK HPR1000 RRI [CCWS], TSP (Ref. 152).

745. The only claim placed on the chemistry regime is that it controls the corrosion to ensure the operational lifetime of the plant. The initial optioneering largely consisted of a review of EPRI guidelines on closed cooling water chemistry choices of the following single chemical inhibitors chromate, hydrazine, hydroxide, molybdate and silica, with the addition of the inhibitor TSP, for which there is Chinese and French OPEX (Ref. 154). I raised two RQs, RQ-UKHPR1000-1078 and RQ-UKHPR1000-1246, requesting further evidence of the relevant OPEX for TSP and also why consideration of additional additives (such as azoles to protect copper-based alloys, required for many reviewed inhibitors), or mixed inhibitor regimes (which can offer advantages over single chemical regimes, such as reduced inhibitor concentration) was omitted from the optioneering (Ref. 3). Whilst the RP’s response only considered the use of azole with TSP rather than with the full range of inhibitors reviewed, it also provided additional support for the use of TSP, noted advantages of operating with simple chemistry regimes (rather than complex multiple additive systems) and updated the ALARP assessment accordingly (Ref. 155). Overall, I am satisfied that the RP has provided sufficient evidence to demonstrate an appropriate RRI [CCWS] chemistry regime to support their claims (Ref. 5). My judgement is based on the RP’s response together with my own knowledge of typical closed cooling water chemistry regimes applied at nuclear power plants and a lack of any reported failures in the industry using TSP (Ref. 156).

746. Clearly, the adequacy of the chemistry regime selected is determined by the materials choices; both are inextricably linked. The SDM sets out the materials selected for the main RRI [CCWS] components, which is mainly carbon steel, except for some use of titanium, copper and stainless steel for valves and heat exchangers (Ref. 150).

747. Justification for these materials choices and compatibility with the selected corrosion inhibitor is provided in ‘Application of Alkaline Agent in the Closed Cooling Water System’ (Ref. 152). Whilst limited detail is given on why certain materials were selected over others, the materials proposed appear reasonable and in line with choices made for similar systems in other PWRs. A suitable justification for the compatibility of the selected corrosion inhibitor with the various component materials is also made, with reference to relevant OPEX from Chinese and French units. I am therefore content with the material and chemistry choices for the RRI [CCWS].

4.4.2.3 Chemistry Control

748. The RP makes the following argument and sub-argument in Chapter 21 of the PCSR for sampling within the RRI [CCWS] (Ref. 5):

- 3.3.10. SC21.10-A1: Sampling and monitoring systems are designed to obtain sufficient and representative sampling of chemical and radiochemical species.
- 3.3.10. SC21.10-A1.5: The radioactive contamination levels in the primary circuit, secondary circuit and auxiliary systems are measured by the PRMS.

749. According to Auxiliary Water Chemistry Specifications, grab samples are taken from a local sampling point on the RRI [CCWS] once per week in order to monitor pH and phosphate concentration (Ref. 147). Monthly grab sampling for iron, calcium, suspended solids, fluoride and chloride is also performed. The limits for both chloride and fluoride (<0.15 mg kg^{-1}) are specified for protection from SCC of the stainless steel components (Ref. 32). Limiting values are also specified for pH ([redacted]), phosphate ([redacted] mg kg^{-1}) and suspended solids ([redacted] mg kg^{-1}). The limits specified appear reasonable and are in line with international guidelines (Ref. 75). No indication of sample point location or any measures in place to ensure representative sampling is provided in the sampled documentation, however there is no reason to suspect that representative chemistry sampling capabilities could not be developed in future for this system. I consider this to be a minor shortfall.

750. The PRMS continuously monitors the radiation level (total gamma) on the outlet of the RRI [CCWS] heat exchanger shell side to identify failures of system heat exchangers cooled by the RRI [CCWS] in a timely manner in order to minimise contamination of the system with radioactive species (Ref. 31, Ref. 152). Once a pre-determined radioactivity level is exceeded, an alarm is triggered, and operators are prompted to carry out surveillance and take any necessary measures to isolate the failed heat exchanger. The RP has set the limit for UK HPR1000 at 0.6 MBq/t based on CGN OPEX, which I consider to be appropriate (Ref. 31).

751. RQ-UKHPR1000-1078 requested details of the RRI [CCWS] interfaces with served systems, including details of the interface (materials, chemistry of the served system) (Ref. 3). The purpose of this RQ was to understand where the main risks of leakage in the RRI [CCWS] exist. The response noted that the main risk of leakage into the RRI [CCWS] is from those interfaces at the greatest differential pressure, which are those associated with the RCP [RCS] thermal barriers and the RCV [CVCS] and REN [NSS] heat exchangers. Detection of such leakage is possible via multiple channels:

- Level measurements in the RRI [CCWS] surge tank. The make-up isolation valve is automatically closed when the level reaches a certain limit. The surge tank is sized to accept such leakage for 30 minutes.
- Temperature or pressure readings and activity monitoring of the heat exchanger outlet by the PRMS. Isolation valves are situated at heat exchanger outlets which will automatically isolate the corresponding heat exchanger in the event of breaching the alarm settings to limit transfer of activity.
- More minor leaks are more likely to be detected in routine chemistry sampling of the RRI [CCWS].

752. Should significant leakage occur, fission gases would be removed from the RRI [CCWS] expansion tanks by the building ventilation systems.

753. The RP has identified a number of safeguards in the design in the event of a leak into the RRI [CCWS]. However, there is a lack of detail of the adequacy of these safeguards, from a chemistry perspective, within the safety case; for example, how much leakage would be expected in the event of a leak, how much a leak would disturb the chemistry and how this compares with the tank capacity. I consider this to be a minor shortfall.

754. Leakage out of the RRI [CCWS] will be detected by level measurements in the RRI [CCWS] or by routine chemical measurements in the served systems. In the event of a small leak, the surge tanks are automatically made up by the NI DWDS. Larger leaks will result in automatic isolation of user systems from the leaking train, before ultimately the pump of that train is stopped.

4.4.2.4 Strengths

755. I did not identify any particular strengths as a result of this part of my assessment.

4.4.2.5 Outcomes

756. Based upon the assessment of RRI [CCWS] chemistry, materials and sampling in UK HPR1000 described in sub-sections 4.4.2.2 and 4.4.2.3 above, I have not identified any Assessment Findings. I have identified two minor shortfalls as discussed in sub-section 4.4.2.3 above.

4.4.2.6 Conclusions

757. Based on the outcome of my assessment of RRI [CCWS] chemistry, materials and sampling in UK HPR1000, and in line with SAPs ECH.1, ECH.3 and ECH.4, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA (Ref. 2).

4.4.3 Radioactive Waste Treatment Systems

758. The generic UK HPR1000 design includes a number of systems that manage radioactive waste to support safe reactor operations. My assessment of these systems began in Step 4 of GDA and was focussed specifically on the adequacy of their design with respect to claims made on the chemical performance required. A broader assessment of these systems was carried out by the Nuclear Liabilities Regulation inspector (Ref. 68).

759. Several sub-arguments of relevance to the chemical performance of the radioactive waste treatment systems are presented in PCSR Chapter 21, including that the gaseous waste system keeps the hydrogen and oxygen concentration in the system and connected components within flammability limits to avoid explosive mixtures (Ref. 5). Additionally, arguments are made that both the liquid and gaseous waste systems minimise radioactive nuclides in wastes to acceptable levels before discharge.

4.4.3.1 Liquid Radioactive Waste

760. The TEU [LWTS] is designed to monitor, collect, store and treat the radioactive liquid waste produced during normal operation of two UK HPR1000 units prior to its discharge to the environment. The system is described in PCSR Chapter 23 and in a dedicated SDM, and the chemistry control aspects are described in PCSR Chapter 21 (Ref. 26, Ref. 157, Ref. 5).

761. The TEU [LWTS] is divided into several sub-systems which deal with waste storage, waste treatment, monitoring and discharge, chemical dosing and sampling and analysis.

762. The NI RPE [VDS] collects effluents from the primary circuit and connected systems which are segregated into recyclable and non-recyclable effluents. The RPE [VDS] coolant storage tanks are swept by the TEG [GWTS] to remove volatile radioactive substances and possible degassed hydrogen for treatment in the TEG [GWTS]. Recyclable effluent is uncontaminated and undiluted reactor coolant and is routed to the TEP [CSTS] for treatment and reuse as supplementary make up for the primary circuit coolant via the REA [RBWMS]. Non-recyclable effluents are segregated into the following streams to facilitate treatment in the TEU [LWTS]:

- Process drains effluents, which are polluted primary coolant from systems or equipment leakage and are unsuitable for reuse. These effluents contain low levels of chemical impurities and are processed by demineralisation.

- Chemical drains effluents, which arise from the radioactive laboratory. These effluents typically have a higher level of chemical impurities and potentially higher radioactivity, and as such are processed by evaporation.
- Floor drains effluents, which arise from leakage, equipment draining and floor washing. These effluents are further segregated into 3 categories based on the level of radioactive contamination expected. They typically contain low levels of radioactive contamination but are high in suspended solids, which are removed by filtration.

763. The liquid waste treatment subsystem includes the demineralisation unit, evaporation unit and filtration unit for carrying out chemical conditioning of the stored effluents. The demineralisation unit consists of three mixed bed demineralisers, a pre-filter and two resin trap filters. The demineralisers are sized to provide sufficient capacity to treat all of the liquid radioactive waste anticipated to be produced by the UK HPR1000 during normal operation. Demineralisation is undertaken on a particular storage tank contents in a closed loop mode until the required discharge concentrations are met. Provisions for resin exchange and transfer are provided in the design. In response to RQ-UKHPR1000-0738, a sizing report which justifies the demineraliser capacity was provided (Ref. 3, Ref. 158). The report explains how key design parameters, including flowrate, resin volume and resin bed capacities have been optimised, taking into account plant data from CPR1000 plants which operate with a similar liquid waste treatment system design.

764. Few details were provided in the PCSR and underlying documentation on the design of the TEU [LWTS] chemical sampling and analysis sub-system, hence RQ-UKHPR1000-0738 was raised (Ref. 5, Ref. 3). Figure 9 was provided in response to the RQ and shows the sub-system manual sampling point locations (red circles), sampling glove boxes (red rectangles) and local sampling funnels (blue circles) which are routed to the glove boxes.

765. In the liquid waste storage subsystem, samples are taken from downstream of the storage tank pumps. Samples will then be transferred to the on-site laboratory to measure the radioactivity and chemical properties of the liquid waste to select the appropriate treatment process. In the liquid waste treatment sub-system, samples are taken downstream of each of the demineralisers and within the evaporation unit to determine the effectiveness of the treatment. Samples are also taken from downstream of the recirculation pump of the evaporation unit to determine the boron concentration, total salinity and sodium-to-boron ratio of the concentrates. Sampling is also performed in the discharge monitoring subsystem after the monitoring tank pump in order to determine whether the chemical and radiochemical composition of the effluent is suitable for discharge.

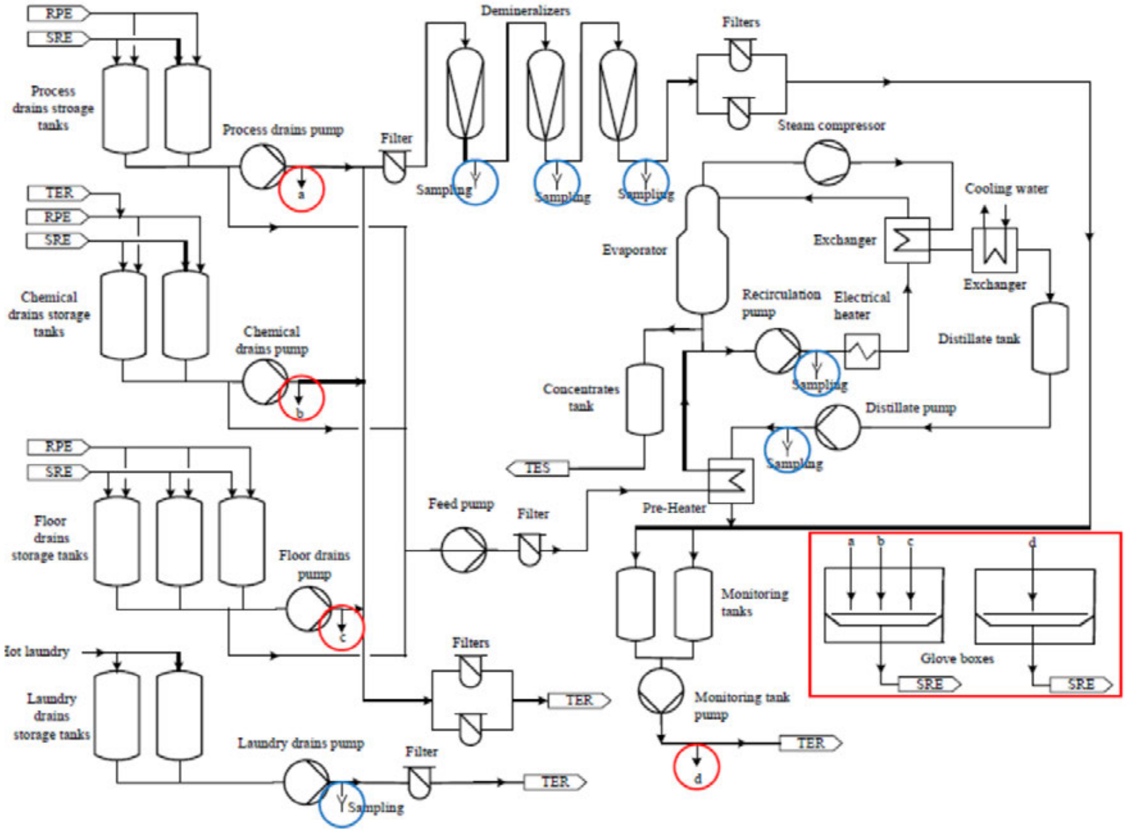

Figure 9: Simplified flow diagram of the TEU [LWTS] highlighting key features of the sampling and analysis sub-system **(Ref. 3)**

766. A particular chemical concern with boric acid solutions is the potential for crystallisation and precipitation. This is generally controlled by maintaining an upper limit in the boron concentration at a given temperature. This is particularly relevant for the evaporator where boron concentrations will increase, and as such, the concentrate tanks and associated pipes are designed with thermal insulation and heat tracing measures to prevent crystallisation of the concentrate.

767. The RP has specified that total gamma, sampled prior to the effluent discharge point, will be a control parameter with limits set in line with the environmental discharge permit. Other TEU [LWTS] limits and conditions necessary in the interests of safety and specific chemistry controls have not been specified at this stage, and will need to be developed; I am content that this can be achieved by the licensee as part of normal business as the design develops.

768. All of the main components of the TEU [LWTS], including the storage tanks, demineraliser vessels and filters, are constructed from stainless steel in order to minimise corrosion.

769. The TEU [LWTS] uses well proven techniques for chemical treatment of the liquid wastes that are similar to other operating PWRs. I am content with the design described from a Chemistry perspective.

4.4.3.2 Gaseous Radioactive Waste

770. The TEG [GWTS] is designed to collect, treat and discharge the gaseous radioactive waste generated under normal operating conditions in one UK HPR1000 unit. Radioactive fission gases, such as krypton and xenon, are produced during normal operations, as well as being released in higher concentrations in the event of fuel cladding failure. The TEG [GWTS] includes a delay unit which holds up these gases

until such time as the radioactivity has decayed to meet limits for discharge to the environment. The TEG [GWTS] also functions to keep the concentration of flammable gases (hydrogen and oxygen) in the TEG [GWTS] and connected components within relevant limits to avoid the presence of explosive mixtures. Consideration of risks due to flammable gases in the system did not form part of my assessment scope, however.

771. The TEG [GWTS] and its chemistry related functions are described in PCSR Chapter 21 and in a SDM (Ref. 5, Ref. 159). Calculations for the sizing of the delay beds and the recombiner were also provided (Ref. 160, Ref. 161). A simplified TEG [GWTS] flow diagram is shown in Figure 10.

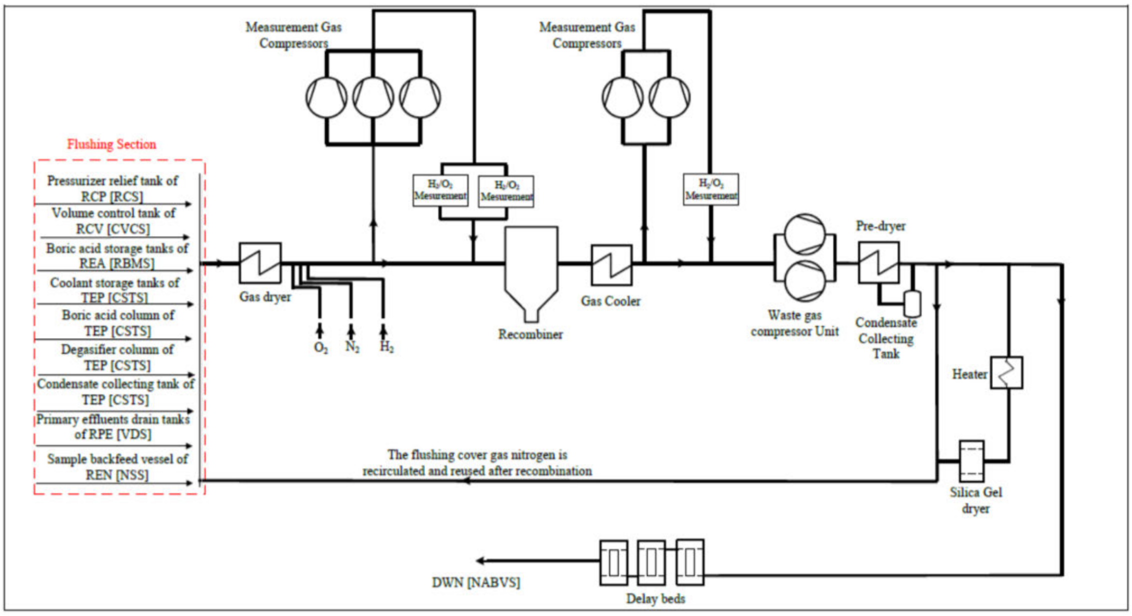

Figure 10: Simplified flow diagram of the TEG [GWTS] **(Ref. 159)**

772. As the radioactive and flammable gases are dissolved in the primary coolant, they are transported to a number of systems within the plant during operations, including the RCV [CVCS] and the TEP [CSTS]. As depicted in Figure 10, the TEG [GWTS] flushes parts of these systems continuously with nitrogen gas to prevent the accumulation of hydrogen. The system also prevents the escape of radioactive gases from the connected components into the atmosphere of the building by maintaining the flushing section at a slight negative pressure, using two redundant waste gas compressors (2 x 100%). As the system operates by continuously recirculating the nitrogen gas, with a proportion removed for treatment by the delay beds, radioactive gases will accumulate within the served systems over time. No limits or conditions have been suggested for the maximum allowable activity in the system at the GDA stage; these will need to be specified by the licensee as part of the normal business development of the safety case.

773. The system consists of pipe connections to the various flushed systems, gas drying equipment, hydrogen and oxygen measurement cabinets, a catalytic recombiner, waste gas compressors and three activated charcoal delay beds. The main components are constructed from stainless steel to minimise corrosion risks.

774. The control of flammable gases is performed by the recombiner, which is a heated pressure vessel filled with catalyst pellets used to convert hydrogen and oxygen from the flushing gas into water. Prior to recombination, the hydrogen and the oxygen concentration in the flushing gas are measured upstream of the recombiner and are used to automatically inject additional hydrogen and oxygen as required in order to maintain a near stoichiometric ratio (slight excess of hydrogen) at the recombiner inlet. When the hydrogen concentration is measured to be higher than 4% by volume and/or

oxygen concentration is measured to be higher than 2% by volume, nitrogen can be injected via the nitrogen injection valves arranged upstream of the recombiner to dilute the flammable gases. The hydrogen concentration at the outlet of the recombiner is designed to be lower than 0.3% and the oxygen concentration lower than 0.1% by volume. The system is sized based on the upper limiting value of dissolved hydrogen in the primary coolant (50 cm^3 kg^{-1}), and on full removal in the connected vessels. The calculation of the sizing of the recombiner is given in a dedicated sizing report (Ref. 161). Whilst the calculations appear conservative, the evidence provided to substantiate the performance of the recombiner is largely theoretical. Further OPEX or other information could be provided to strengthen the justification; I consider this to be a minor shortfall. The RP's safety classification of the recombiner is F-SC3; I consider this to be appropriate from a Chemistry perspective, given the chemistry claims on the system.

775. Conditioning of the radioactive component of the gaseous waste prior to discharge is performed by the activated charcoal delay beds. The delay beds are sized to allow for a minimum of 40 hours decay for krypton isotopes and 40 days decay for xenon isotopes. Krypton and xenon are retained on the bed by a reversible dynamic physical adsorption process which does not consume the charcoal. The delay time is affected by factors including the characteristics and moisture content of the activated charcoal, and the operating temperature and pressure of the delay beds. A sizing report was produced by the RP to justify the mass of charcoal in the three beds (Ref. 160). The sizing calculation is based on operating conditions which are expected to generate a large gaseous waste flow rate, such as nitrogen flushing of the primary loop and pressuriser, and a large safety margin is added (the charcoal mass required to achieve the selected delay times was calculated to be 0.91 tons, however 7.2 tons was selected, in line with FCG3 and OPEX from similar designs).

776. Since moisture can affect the adsorption of the noble gases on the charcoal, a pre-dryer is situated upstream of the delay beds to reduce the relative humidity of the incoming gas, and a silica gel dryer is situated downstream of the pre-dryer to further dry the gas before it enters into the delay beds. Two sets of humidity meters are also situated prior to the delay beds; if high humidity is detected, the delay beds will be automatically isolated, and the waste gas is returned to the TEG [GWTS] flushing section.

777. The UK HPR1000 TEG [GWTS] uses well proven techniques for the treatment of gaseous wastes, that are similar to other operating PWRs. The system capacity calculations appear conservative and, overall, I am content with the design described from a Chemistry perspective.

4.4.3.3 Strengths

778. The RP has provided a coherent, adequate safety case to justify the chemistry related functions of the radioactive waste systems at an appropriate level within the safety case.

4.4.3.4 Outcomes

779. Based upon the assessment of the liquid and gaseous radioactive waste treatment systems in UK HPR1000 described in sub-sections 4.4.3 above, I have not identified any Assessment Findings. I have identified a minor shortfall, as described in sub-section 4.4.3.2.

4.4.3.5 Conclusions

780. Based on the outcome of my assessment of the TEU [LWTS] and TEG [GWTS] in UK HPR1000, I have concluded that that the claims, arguments and evidence as

presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

4.5 Accident Chemistry

781. The safety case for a nuclear reactor should consider the risks arising during both normal operations, and from fault/accident conditions. Accidents can be minimised by the implementation of a conservative design, through good operational practice, and by the design of an adequate maintenance and testing regime. Nuclear facilities should therefore be designed to cope with, or withstand, a wide range of faults, whilst not incurring unacceptable consequences, through a combination of the inherent design, and identified safety measures.

782. A subset of faults or accidents involve chemistry, either through instigation of the fault, or as part of the resultant accident source term. An adequate safety case should describe how chemical hazards are understood and minimised, and that appropriate assumptions have been made on the various assessments that the RP has undertaken. Chemistry is a factor in both Design Basis Analysis accidents, named as DBC accidents in the RP's case, as well as Beyond Design Basis faults, and Severe Accident Analysis, which are named as Design Extension Condition (DEC) accidents in the RP's case, (DEC-A and DEC-B respectively). I have therefore sampled a number of faults from both DBC and DEC categories, to provide confidence in the overall approach taken by the RP.

783. I engaged a TSC to support me in my assessment of accident chemistry, and the assessment was performed during both Step 3 and Step 4 of the GDA. I also collaborated with colleagues from Fault Studies and Severe Accident Analysis, where appropriate. During Step 3 of GDA, my TSC produced a document that set out their understanding of RGP in the accident chemistry topic, as well as a high level overview of the RP's generic safety case (Ref. 162, Ref. 163). The RGP report was then used as the basis for the more detailed assessment that followed in Step 4 of GDA (Ref. 164, Ref. 165, Ref. 166).

4.5.1 Overview

784. PWRs have three successive barriers to prevent radioactivity release during an accident:

- Fuel cladding
- Primary circuit
- Containment

785. During normal operations, the fuel cladding will provide the primary means of preventing the release of radioactivity and should retain over 99% of the nuclear material in the reactor. Should the fuel cladding fail, radioactivity will enter the primary circuit, from where it can be released into the containment, and ultimately the environment if the containment fails or leaks.

786. The UK HPR1000 has several safety systems whose functions are to control reactivity or remove heat. Of relevance to accident chemistry, are the following systems (Ref. 14):

- The RIS [SIS] provides borated water injection into the RCP [RCS] to compensate for the water inventory loss under certain faults. In addition, the RIS [SIS] can operate in RHR mode, performing residual heat removal functions in the long-term after an accident.

- The RBS [EBS] injects highly borated water into the RCP [RCS] via the RIS [SIS] cold leg injection line, to control the reactivity of the reactor during the transition from the controlled state to the safe state.
- The ASG [EFWS] provides emergency feedwater for the SGs to remove residual heat when the normal feedwater systems are unavailable during a range of faults.
- The EHR [CHRS] provides a diverse means of containment protection, by removing heat from the containment with containment sprays to maintain containment integrity, and by injecting water to the reactor pit and cooling the IRWST.
- The Containment Filtration and Exhaust System (EUF [CFES]) performs an active pressure relief function during DEC scenarios.
- The Containment Combustible Gas Control System (EUH [CCGCS]) controls the hydrogen concentration in the containment.

The safety functions of these systems are assessed elsewhere (Ref. 117, Ref. 167).

787. I chose to sample a range of phenomena related chemistry, that span both DBC and DEC faults.

788. The RP has included two main sub-claims regarding accident chemistry:

- 3.3.10.SC21.8: The chemistry and process are optimised in fault and accident to maintain the integrity of the safety barriers.
- 3.3.10.SC21.9: Radioactivity level arising from accident conditions in the UK HPR1000 is reduced SFAIRP to minimise worker and public dose.

789. Several sub-claims are also made under these claims, and these are described in the following sections.

790. The chemistry PCSR includes a summary of the main chemistry assumptions and analysis that have been used in the wider safety analysis, which are described in PCSR Chapter 12 and Chapter 13 (Ref. 5, Ref. 168, Ref. 23). These have been assessed by Fault Studies and Severe Accident Analysis respectively (Ref. 117, Ref. 167).

791. During my assessment of accident chemistry, I used the guidance set out in the SAPs (Ref. 2). In particular, I used SAPs ECH.1, ECH.2 and ECH.3, which require an adequate safety case to be produced which considers conflicting chemical effects, and demonstrates that chemistry can be adequately controlled. I used SAPs FA.1 and FA.15, which set out general requirements for Design Basis and Severe Accident Analysis, respectively, and their corresponding TAGs NS-TAST-GD-006 and NS-TAST-GD-007. I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment, which consider the effect chemistry has in accidents (Ref. 6).

4.5.2 Control of Safety System Chemistry

792. The UK HPR1000 generic design features a number of systems that are used to maintain the safe operation of the plant. Some of these systems are needed to maintain the required concentration of boron and to remove heat, both during normal operations and accident scenarios. The control of the chemistry of these systems is a key requirement, and should correlate with that of the primary circuit, both in terms of boron concentration and impurities.

793. The main relevant systems in the generic design of the UK HPR1000 are as follows (Ref. 5):

- Boric acid is made-up and stored in the REA [RBWMS], which is part of the RCV [CVCS].
- Concentrated boric acid can be supplied during accidents by the RBS [EBS].
- A further source of boric acid for supply during accidents is available from the RIS [SIS]. This system also has a role in pH control of the IRWST, during accidents, via the addition of TSP.
- A further source of borated water is the IRWST, which stores borated water to serve as a water source during safety injections.

794. The REA [RBWMS] is required to provide the initial filling of borated water and make-up for the primary circuit and other boron-bearing systems from start-up. It also controls the boron concentration in the primary circuit during normal operations, to control the slow reactivity change in the core, and tracks the expected load changes, by increasing or decreasing the boron concentration. This system was assessed in detail in sub-section 4.2.2.2 of this report alongside the RCV [CVCS], and these systems will not be discussed further in this section.

795. This sub-section also considers chemistry control in the Emergency Feedwater System (ASG [EFWS]), which has the function of removing residual heat from the SGs when other feedwater systems are unavailable. Also of relevance to safety system chemistry, and assessed here, is the Containment Filtration and Exhaust System EUF [CFES] which has the function of maintaining confinement of radioactivity following a DEC accident.

4.5.2.1 Emergency Boration System

796. The RBS [EBS] is composed of three identical independent trains, each of which is designed to have 100% capacity for emergency boration (Ref. 20, Ref. 128). Each train consists of one emergency boric acid tank and one pump, and has trace heating. Borated water is injected into the cold leg of the corresponding loop of the primary circuit via a connection line provided in the RIS [SIS]. The Safety Functional Requirements of the RBS [EBS] are provided in the PCSR, with the RBS [EBS] required to provide borated water under certain fault conditions (Ref. 20).

797. As part of a contract to assess the safety case concerned with the control of boron chemistry, I asked my TSC to provide a high-level review of the RBS [EBS] safety case, which included the Safety Functional Requirements (SFR) of the RBS [EBS], and other relevant documentation, which contributed to my assessment (Ref. 169).

798. Each train of the RBS [EBS] contains water borated with EBA (enriched at 35 at%) with a minimum concentration of 7000 mg kg^{-1}, identical to minimum concentration as contained within the BASTs in the REA [RBWMS]. In response to RQ-UKHPR1000-1221, the RP provided an adequate justification of the volume of the RBS [EBS] storage tanks, which gave me confidence that the RBS [EBS] had been sized appropriately to adequately meet the requirements set out in the SFR.

4.5.2.2 Safety Injection System, Residual Heat Removal System and In-Containment Refuelling Water Storage Tank

799. The RIS [SIS]/RHRS is described in the PCSR as a dual-purpose system, that is both a source of borated water, available for injection during both DBCs and DECs, as well as a residual heat removal system via heat exchangers (Ref. 20). The RIS [SIS]/RHRS consists of three independent trains, each located in a separate safeguard building. When in RIS [SIS] mode, each train can inject borated water via a pump, accumulator and heat exchanger, via the RIS [SIS] cold injection line.

800. Under normal shutdown states, the RIS [SIS] operates in the RHR mode and performs the function of residual heat removal, and the RIS [SIS] cools the primary coolant loop

to the cold shutdown state by operating the RHR heat exchanger. The RHR mode also has a role in certain fault/accident conditions; when the SGs cannot effectively remove the core decay heat, the RHR heat exchanger in the RIS [SIS] can perform this function. The RIS [SIS] takes borated water from the IRWST, to inject into the primary coolant during an accident. The IRWST is an open structure and is constructed with a stainless steel liner, located in the bottom of the reactor building. The water inventory has been designed to be sufficient to provide water to the RIS [SIS] and the EHR [CHRS], (see sub-section 4.5.1), following an accident. The boron concentration of the water inside the IRWST is 1300-1400 mg kg^{-1}, and like the other systems, is enriched in ^{10}B to 35%. However, no consideration has been made about the possibility of contamination or radioactivity being fed forwards upon the start of RHR mode, from the IRWST. I consider this to be a minor shortfall.

801. In order to ensure the reliable operation of the RIS [SIS] and EHR [CHRS] pumps after an accident, the IRWST is equipped with a filtering system which intercepts and filters any debris washed into the IRWST. An alkaline environment is required to be maintained in the IRWST post-accident, and to achieve this, the IRWST has pH adjustment baskets containing TSP. The basket is placed in the BRX and in the water flow path to the IRWST after the occurrence of a Loss of Coolant Accident (LOCA), and accidents where containment spray is required. In the case of a LOCA, the borated water flowing from the break dissolves the TSP granules in the basket and enters into the IRWST. The TSP is designed to maintain the pH value of the water inside the IRWST above 7.0. This chemical characteristic of the water can keep the radionuclides (particularly the iodine) in the sump water of the containment, so that the radioactive iodine cannot escape from the solution long after the accident occurs. This aspect of the safety case is considered in more detail in sub-section 4.5.4.

802. Clearly, the control of both the concentration and enrichment of boron in these systems is of fundamental importance in ensuring that the Safety Functional Requirements are delivered. This aspect of my assessment can be found in sub-section 4.2.2.

4.5.2.3 Emergency Feedwater System

803. The ASG [EFWS] supplies water to the SGs to remove residual heat in the event that the ARE [MFFCS] and the AAD [SSFS] are unavailable. The system is in a standby state during normal operations but may be used during shutdown to supply water (chemically dosed) for the wet lay-up of the SGs, and can be used to supply water to the SGs during commissioning and plant start-up.

804. The ASG [EFWS] is not described in PCSR Chapter 21, although the system is referred to in PCSR Chapter 11 and described in a SDM (Ref. 22, Ref. 170). The ASG [EFWS] consists of three identical trains corresponding to each SG. It is a simple system with each train consisting of a storage tank, an emergency feedwater pump, a flow limitation control valve, a SG level control valve and containment isolation valves.

805. The Conventional Island Demineralised Water Distribution System (SER [DWDS (CI)]) supplies demineralised water to the ASG [EFWS] storage tanks and the Secondary Passive Heat Removal System (SPHRS [ASP]) can also supply water should the SER [DWDS CI)] be unavailable. The SER [DWDS CI)] supplies demineralised water that has been treated to achieve a target pH of [redacted] and to minimise impurities (Ref. 134). Because the water stored in the ASG [EFWS] tank is aerated and can contain elevated levels of dissolved oxygen, the ASG [EFWS] is largely constructed from stainless steel to minimise corrosion. The NI Chemical Reagents Distribution System (CDS [SIH]) injects ammonia and hydrazine at the discharge of the emergency feedwater pumps to achieve an appropriate water quality for use during SG wet lay-up (SG lay-up is assessed in sub-section 4.3.6).

806. Whilst the supply of water from the ASG [EFWS] to the SGs during SG lay-up is chemically dosed (with ammonia and hydrazine) to maintain a reducing environment, consideration does not appear to have been given to the conditioning of ASG [EFWS] water to minimise dissolved oxygen levels prior to supply to the SGs during fault/accident conditions. Although the use of the system under these circumstances is expected to be infrequent, the introduction of oxygenated feedwater at high temperatures could lead to corrosion of the SGs. Other PWRs have used stored condensate as a source of emergency feedwater, with additional measures to minimise pick-up of dissolved oxygen such as nitrogen blanketing. The generic safety case also does not appear to consider such measures. If the intent is not to condition ASG [EFWS] water to minimise dissolved oxygen levels prior to supply to the SGs, I consider that further justification of this design decision is required, including consideration of any appropriate mitigations. I consider this to be an Assessment Finding.

AF-UKHPR1000-0160 – The licensee shall justify the consequences of introducing feedwater with elevated dissolved oxygen levels into the steam generators, from the emergency feedwater system.

807. Chemical specifications for ASG [EFWS] water are detailed in the secondary circuit water quality specification (Ref. 134). However, no capability for sampling and monitoring of the system is described in the Feedwater Chemical Sampling System (SIT [FCSS]) DSR, the relevant SDMs, nor in the rest of the chemistry sampling and monitoring documentation (Ref. 143). An overview of provisions for sampling and monitoring in the ASG [EFWS], including justification of the sampling system design, should be provided as the design develops in the site-specific stages. I consider this to be a minor shortfall.

4.5.2.4 Containment Filtration and Exhaust System

808. The function of the EUF [CFES] is to maintain confinement of radioactivity following a DEC accident by providing pressure relief and therefore maintaining containment integrity. The filter unit is intended to keep the quantity of radioactive substances released into the environment as low as possible in the course of pressure release. The design basis of the system is for continuous operation for 12 hours with a venting mass flow rate at 0.52 MPa (a) of at least 4 kg s^{-1}. The filtration efficiencies are 99.9% for aerosols, 99% for diatomic iodine (I_2), and 80% for organic iodine (Ref. 171). After operating for 12 hours, the system needs to be shut down in order to replenish the chemical reagents.

Design

809. The system is described as a line from containment entering the EUF [CFES] via two manually actuated isolation valves in series. The feed then enters a combined filter unit, where particulates and gaseous forms of iodine are retained via a mechanical filter and presumably through a gas bubbler. However, precise details of the design, and of the filtration unit chemistry (including the need to shut the system down after 12 hours to replenish the chemicals) were not provided in the safety case. The limited specification does state that the exhaust from the filter unit passes through a restriction orifice which regulates the gas flow rate through the filter and from the exhaust. Downstream is a rupture disc with a setpoint of 0.08 MPa (g) (Ref. 172). The system is also equipped with radiation monitoring equipment which is for the purpose of minimising radiation release. Upon receipt of a low level alarm from the radioactivity monitor, the EUF [CFES] is suspended, and liquid make-up is carried out on the system before restarting. The lack of detail about the design of the EUF [CFES] does not meet my expectations for this area of the safety case, and therefore I consider this to be an Assessment Finding.

AF-UKHPR1000-0170 – The licensee shall, as part of detailed design, specify and justify the design of the containment filtration and exhaust system. This should include, but not be limited to:

- defining all chemical requirements;
- defining appropriate limits and conditions;
- demonstrating that the design is capable of mitigating releases of radiation during operation of the unit;
- demonstrating that the risk due to hydrogen combustion during accident scenarios has been mitigated;
- justifying that the chemical effects relevant to the operation of the system have been considered as part of defining an adequate source term; and
- demonstrating that risks have been reduced so far as is reasonably practicable.

810. The system is kept under nitrogen while on standby, but there is no special requirement for an inert atmosphere during operation according to the filtration performance requirements. The safety case does not state the purpose of inerting the system. Although the RP stated in the response to RQ-UKHPR1000-1476 that the inert atmosphere was to keep the filter chemicals in a steady state over long periods of standby, to prevent carbonation of an alkaline reagent, the RP did not clarify the requirements for additional reagents (Ref. 3). Nor did the RP consider the need for inerting to mitigate the risks from hydrogen combustion during accident scenarios. I consider this to be part of Assessment Finding AF-UKHPR1000-0170.

811. The filter itself is most effective for iodine and aerosols, and less so for organic iodine. Therefore, the possibility of formation of organic iodine upstream of the filter should be considered during material selection with regards to any paints or polymer components.

812. According to the SDM of the EUF [CFES], clogging of the system by aerosols will be prevented, but there is no discussion of how this will be achieved (Ref. 172). Considering that the EUF [CFES] is likely to be initiated due to loss of containment sprays, this implies that the aerosol levels in the containment gas mixture could be elevated. Detailed consideration should be given to how the system will be able to manage the expected aerosol loading through the lines and into the filter. This includes flow through structures such as the isolation valves, which may cause flow perturbations and encourage deposition. As the design is not yet finalised, these two aspects should be reviewed again at a later stage when the component information has been made available. I consider this to be part of Assessment Finding AF-UKHPR1000-0170 described above.

813. The unavailability of containment sprays will result in a lack of pH control within containment because they normally wash aerosols (which contain fission products including iodine and caesium compounds, and ruthenium species if the core is exposed to oxidising conditions) and gaseous iodine (I_2, RI) out of the atmosphere and into the alkaline sump where retention as iodide ions is facilitated and radiolytic formation of volatile iodine is suppressed. Without the sprays, aerosol-borne iodine may deposit in unbuffered water films and droplets and undergo conversion to I_2, (Ref. 163, Ref. 162). Since gaseous iodine is less efficiently retained by filters than iodine in aerosol form, mechanisms which enhance the former will have an impact on the radiological consequences. The situation would be exacerbated by the radiolytic formation of nitric acid from the humid air in containment.

814. The iodine chemistry of aerosols and gases in containment should be properly considered when determining the accident source term, since use of a source term assuming sprays are operational would not be appropriate or bounding. Nor would

assuming that all of the aerosol remains suspended and unchanged in the containment atmosphere throughout the accident (Ref. 162, Ref. 163). In response to RQ-UKHPR1000-1476, the RP stated that the quantity of suspended aerosol will not be significant due to settling, and that the water volumes in containment will be assumed to be acidic except for the IRWST which would be under pH control (Ref. 3). As detailed in sub-section 4.5.4 of this report, iodine chemistry in water films and in the reactor pit is not considered, and a more recent assessment by the RP suggests that pH control would be lost from the IRWST early in the accident (Ref. 164). This is part of Assessment Finding AF-UKHPR1000-0168, which is raised in sub-section 4.5.4.1.

815. Regarding the water volumes outside of the IRWST, the reactor pit can be regarded as alkaline, since the EUF [CFES] operates some time into the accident, by which point the water from the alkaline IRWST would have been transferred to the reactor pit. The water films however should be regarded as acidic, because EUF [CFES] is used in the absence of sprays. Therefore, alkaline water will not be recirculated from the sump onto containment surfaces, and radiolytic formation of nitric acid would acidify any unbuffered water films. I consider this to be part of Assessment Finding AF-UKHPR1000-0170**.**

Combined Filter Unit

816. There is a lack of detail presented in the safety case about the filter unit, as a detailed design has not yet been completed (Ref. 172). The documentation refers to chemicals which can fulfil the function of reducing radioactive emissions, but these are not specified beyond stating they will be alkaline, as set out in the response to RQ-UKHPR1000-1476 (Ref. 3). This is related to Assessment Finding AF-UKHPR1000-0148. I am therefore unable to perform any kind of meaningful assessment on the filter unit.

817. Despite the lack of a detailed design, the RP has described some specifications for filtration performance, and these require good efficiencies for aerosols and gaseous inorganic iodine (up to 99.9%), and lesser retention of organic iodine (80%). I am aware that systems similar to the EUF [CFES], (also known as Filtered Containment Venting System in the wider nuclear industry), have been incorporated within the designs of many nuclear power plants, so I have some confidence that such a system could be successfully incorporated into the UK HPR1000. I am also aware of containment filtration units that have been shown to achieve efficiencies of around 98% for organic iodine (Ref. 166).

4.5.2.5 Strengths

818. The RP has provided a coherent, adequate safety case to justify the chemistry functions of relevant safety systems at an appropriate level within the safety case. The EUF [CFES] could, in principle, be an effective mitigation device if it is adequately substantiated in a future safety case.

4.5.2.6 Outcomes

819. Based upon the assessment of the control of safety systems chemistry in UK HPR1000 described in sub-section 4.5.2 above, I have identified two Assessment Findings which need to be addressed by the licensee. These concern the consequences of introducing feedwater with elevated dissolved oxygen levels into the steam generators from the emergency feedwater system (AF-UKHPR1000-0160), and the specification and justification of the design of the EUF [CFES] (AF-UKHPR1000-0170). Part of AF-UKHPR1000-0168 is also discussed in this section of the report. I raised two minor shortfalls.

4.5.2.7 Conclusion

820. Based on the outcome of my assessment of the control of safety systems chemistry in UK HPR1000, I have concluded that that in general the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA. Regarding my assessment of the EUF [CFES] in UK HPR1000, I have concluded that that although important details of the operating chemistry are yet to be defined, the overall approach taken by the RP should be achievable, and therefore the RP has made an adequate case to support GDA.

821. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used the guidance contained in SAPs FA.1, which required severe accident analysis to be performed to demonstrate that risks are reduced SFAIRP. I used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment, which provide guidance about the expectations for safety cases of civil nuclear reactors (Ref. 6) and NS-TAST-GD-006 and 007, which concern Design Basis Analysis and Severe Accident Analysis (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the generic safety case.

4.5.3 Chemistry Assumptions in Design Basis Condition Analysis

822. The RP produced a suite of information regarding DBCs for the UK HPR1000 (Ref. 173, Ref. 174, Ref. 175, Ref. 176, Ref. 177, Ref. 178). The ‘Design Basis Accident Source Term Methodology Report’ expands on the relevant chemical phenomena of the DBCs (Ref. 173). The faults described and treated by the RP and of relevance to this assessment are (Ref. 173):

- Large Break – Loss of Coolant Accident (LB-LOCA)
- Dropping of Fuel Assembly
- Steam Generator Tube Rupture (SGTR) (One tube)
- SGTR (Two Tubes)
- Small Break – Loss of Coolant Accident (SB-LOCA)

823. SGTR faults, due to their particular sensitivity to chemistry assumptions, have been treated separately throughout this section of my assessment. The methodology employed by the RP throughout the DBC and SGTR source term submissions is based heavily on guidance published by US Nuclear Regulatory Commission, specifically NUREG-1465 ‘Accident Source Terms for Light-Water Nuclear Power Plants’, and Regulatory Guide (RG) 1.183 ‘Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors’ (Ref. 179, Ref. 180). The Regulatory Guide is intended as guidance for US licensees on evaluating the radiological consequences of DBCs in Light Water Reactors (LWRs).

824. The key chemistry aspects described in the RP’s submissions, are as follows:

- Core inventory
- Primary coolant activity
- Transfer of radionuclides from the primary side to the secondary side
- Iodine chemical form
- Activity release from discharged liquid
- Retention of radioactivity by natural and engineered processes
- Transfer of radioactivity into the steam space in a SG

4.5.3.1 Source Terms

825. Detailed source term methodologies are available for some of the accidents in the submissions made by the RP, either in the appendices to DBC Source Term Analysis document (Ref. 173), or for the LOCA and SGTR faults in additional reports (Ref. 175, Ref. 176, Ref. 177, Ref. 178). In many cases the details of the radionuclide sources and radioactivity transport are common to some of the other faults too. I decided to sample a selection of the accidents to gain confidence about how the source term analysis had been produced.

826. The RP has defined release fractions and the timings of releases in accordance with RG 1.183, which is based upon NUREG-1465 (Ref. 180, Ref. 179). All radionuclides are assumed to be in particulate form, with the exception of noble gases, iodine, and organic iodides (Ref. 173). As further clarified during Step 3 of GDA, in Design Basis Faults, where releases are associated with leaking coolant, practically all of the non-volatiles will be in ionic form and dissolved in water droplets (aerosol) or in the bulk water (Ref. 163). The RP's 'Design Basis Analysis Source Term Methodology', states "During the radioactivity transport progression, the chemical form may be changed and will be analysed in detail combined with the specific accident" (Ref. 173). However, this level of detail is absent in most of the accident analyses included in the appendices, and also in the separate assessments (Ref. 175, Ref. 176, Ref. 177, Ref. 178). My assessment of the approach taken by the RP is discussed in subsequent sections of this report. The chemical form of noble gases will not change as the faults proceed, but the chemical form of iodine and caesium will depend on the fault/accident conditions.

Loss of Coolant Accidents

827. In the event of a LOCA, the RIS [SIS], is activated. As part of this system, water from the LOCA follows a flow path to the IRWST, through one or more baskets of TSP, a salt that when dissolved in water produces an alkaline pH at low concentrations.

828. For the UK HPR1000, the hydrated salt will be used, and around 5.6 te will be used which is approximately a quarter of the effective TSP mass present in the Sizewell B containment, (Ref. 163). During an accident such as a LOCA, radiolytic reactions and carbon dioxide dissolution within the containment will affect the pH balance of the aqueous system by the production of acids and bases. The RP has presented calculations that show that a pH >7.0 would be maintained for around 30 days following the LOCA (Ref. 181). The assumptions made by the RP are largely reasonable, and radiolytic nitric acid (HNO_3) production within the containment atmosphere will largely determine the longer-term pH levels in the IRWST. I am content that the buffering capacity of the IRWST is adequate for DBCs, and that the amount of TSP described by the RP is adequate.

829. Hydrated TSP is highly soluble in water and has a low melting point of 73.4 °C. The high ambient temperature during either DBCs or DECs could cause the TSP to start melting, causing aggregation and slowing down dissolution. Salts such as TSP may be prone to caking or clumping with prolonged storage, which might also adversely affect the solubility. Considering that the quantity of TSP in containment would be far less than is available for the existing UK PWR at Sizewell B, I asked the RP to demonstrate that conditions at the TSP baskets would not adversely impact the solubility such that it would not be able to dissolve sufficiently quickly to provide the necessary pH levels when required. The RP confirmed that the ambient temperature of the TSP in its location is controlled between 15-45 °C. Each refuelling outage, the condition of the TSP will be checked, and the solubility and pH effect of the salt will be tested. If the testing shows that the TSP is degrading, the chemical will be replaced. The RP also demonstrated that the TSP basket is located within a pit in the flow path to the IRWST in the response to RQ-UKHPR1000-1353 (Ref. 3). This is designed so that even if the

TSP were to relocate due to deliquescence or any other mechanism then the TSP would still be transported to the IRWST. I was satisfied with these arrangements.

Dropped Fuel Assembly

830. The postulated dropped fuel fault occurs outside of the containment building, and therefore stands alone against the other faults in terms of the plant and systems affected. A Fuel Handling Accident results in a fuel assembly being damaged during fuel handling operations at the SFP. Damaged fuel rods are assumed to immediately release the gap inventory to the water in the pool. Because there is no loss of cooling to the fuel, only the gap inventory is released. The RP states that this comprises noble gases, halogens (including iodine), caesium, and rubidium, and that caesium and rubidium are assumed to be released from the gap inventory and dissolve fully in the water of the SFP (Ref. 173, Ref. 174).

831. The RP assumes that the iodine is released from the fuel as 95% caesium iodide (CsI), 4.85% elemental iodine (I_2), and 0.15% organic iodine, which is the same as the chemical form assumptions made for radioactivity released from the RCP [RCS] to containment (Ref. 173). The analysis states that the low pH of the fuel pool causes caesium iodide to transform to I_2 in the pool water. The assumption is made that all of the caesium iodide is instantaneously converted to I_2. Therefore, the species in the pool are 99.85% I_2, and 0.15% organic iodine. I am content that this is a conservative assumption, as not all of the caesium iodide will instantaneously convert to I_2 (Ref. 165).

832. The methodology then assumes that the SFP provides a Decontamination Factor (DF) of 500 for the I_2 evolved in the SFP, and a DF of one for organic iodine. No retention in the SFP is claimed for noble gases, which is appropriate. The DF of 500 is taken directly from RG 1.183, with the caveat that this should only apply when the depth of water above the damaged fuel exceeds seven metres; the RP states that the depth of water to be 7.8 metres (Ref. 180, Ref. 173). The RP also confirmed that should a fuel assembly come to rest higher than the bottom of the SFP (on a fuel rack, for example), this depth would still be maintained.

833. A DF of 500 in practice means that 99.8% of the I_2 is retained in the water of the SFP. The acidic pH in the pool will favour the radiolytic formation of I_2 and atomic iodine, though these processes would become less important with distance from the fuel failure as the radionuclides are carried away from the fuel by convection and diffusion. Although the RP did not provide a chemical basis for their approach (instead referencing RG 1.183), my TSC was content that this is a conservative assumption, and therefore I am content with the approach taken by the RP (Ref. 180).

834. The containment building is served by the Containment Sweeping and Blowdown Ventilation System early in the fault, then ventilation is switched over to one of the safety trains of the Annulus Ventilation System, by the reactor protection system. In the event of a dropped fuel assembly accident within the fuel building, the BFX Ventilation System, operates for the first 30 minutes of the accident. No filtration of radionuclides is claimed in this ventilation mode. After 30 minutes, the Safeguard Building Controlled Area Ventilation System, is initiated. The RP has provided the following filtration efficiencies in Table 15 (Ref. 173):

Table 15: Filtration Efficiencies for Various Radionuclides

Radionuclide Type	Filtration Efficiency
Noble Gases	0%
Organic Iodine	99%
I_2	99%
Aerosols (including caesium iodide and other caesium species)	99.9%

835. The RP describes these as conservative efficiencies, as in reality, better performance is expected. The RP provided clarification in the responses to RQ-UKHPR1000-0659 and RQ-UKHPR1000-1353, of the applicability of this information to the UK HPR1000 (Ref. 3). The response included OPEX from CGN plants, where the requirements in their technical specifications are 99.99% for HEPA filters and 99.9% for iodine absorbers. The RP also referenced to a requirement to comply with current UK practice (as referenced by the RP to a Sellafield Engineering Standard), to ensure HEPA filtration efficiency of 99.9925% for particulates (Ref. 182). My TSC indicated that these values are likely to be conservative, and I am therefore content with the approach taken by the RP (Ref. 165).

Steam Generator Tube Rupture

836. In an SGTR, all of the noble gas activity entering the secondary side of the SG is assumed to be released directly to the environment. A fraction of the iodine and alkali metal activity in the SG water is carried over into the steam phase. The carryover fraction, or partition factors, are justified by the RP, and are a simplified version of RG 1.183 (Ref. 173, Ref. 180). For alkali metals, the carryover fraction of 0.0025 is equal to the maximum steam moisture content at the outlet for the SGs under normal operation. This should be conservative since the moisture content is expected to be considerably lower than this design value, and so this is considered by the RP to be a bounding value during normal operation.

837. The iodine partition coefficient (the mass of I_2 per unit mass of liquid water/mass I_2 per unit mass of steam) is taken to be 100, which gives a release fraction of 0.01, indicating that most of the iodine remains in the liquid water. The RP states that all of the release is conservatively assumed to be I_2 in the radiological consequences assessment. As a basis for the SGTR source term, I am content that these assumptions are conservative, a conclusion which is supported by the advice provided by my TSC (Ref. 164).

838. Two SGTR faults are considered by the RP (Ref. 173):

- SGTR (one tube)
- SGTR (two tubes in one SG)

839. The analysis for these accidents was provided by the RP in two reports (Ref. 177, Ref. 178). The relevant source term for this accident is the primary coolant activity. During Step 4 of GDA, the RP's safety case for SGTR faults included a number of chemistry assumptions, which were assessed by my TSC. The main assumptions were as follows:

- The reactor is operating with a limited number of fuel failures at the time of SGTR, which gives an initial coolant activity of the operating limit of 16 GBq t^{-1},
- The concurrent iodine spike model is applied, resulting in a 335x increase in the release rate of iodine from the defective fuel,
- All iodine is present as I_2.

840. The spiking model used assumes a spike release rate 335 times greater than the equilibrium release rate for the SGTR accidents. The RP clarified in the response to RQ-UKHPR1000-1353 that only iodine is taken into account in this model (Ref. 3). The RP stated that the sole use of iodine in concurrent spiking (excluding noble gases and alkali metals, for example) was in accordance with a document prepared by the European Commission Reactor Safety Working Group, which compares the approaches adopted across several European countries (Ref. 183). The European Commission study does consider spiking of caesium, with factors ranging from 30-115 (Ref. 183). The RP states that the methodology of RG 1.183, while not including caesium, has a much higher iodine spiking factor of 335 for SGTR and is therefore conservative (Ref. 180). I am content that this approach is reasonable, and this is also supported by independent advice provided by my TSC (Ref. 165).

841. Although I am content that this is a conservative approach, my TSC indicated that the RP’s approach (using RG 1.183) was likely to be more conservative than other possible methods (Ref. 165). Following my initial assessment, Fault Studies colleagues indicated that the UK consequence analysis was undertaken with the assumption of food bans, and that this was not appropriate. As a result, the RP committed to reviewing the possible options for removing some of the conservatisms that had been included in the original safety case (Ref. 184). The approach adopted by the RP included changes to the assumptions made about chemistry. As well as the production of the optioneering study, the RP also updated the SGTR assessments (Ref. 184, Ref. 175, Ref. 176). In the new analysis, the RP made the following changes to the assumptions regarding iodine releases from the SGTR:

- The iodine speciation in the RCP [RCS] is assumed to be the involatile iodide (I^-) species, with only a very small amount (0.1%) of volatile iodine; this speciation is also assumed for iodine passed into the SG. The original assessment assumed the chemical form of iodine in both the RCP [RCS] and the SG to be I_2.
- The new analysis assumes all the I_2 in the SG is released directly into the steam, as for noble gases. The original assessment assumed a partition factor for I_2 of 0.01.
- The new analysis assumes that iodide is released into the steam phase of the SG with a partition factor of 0.0025, as is assumed for particulate fission products.
- The primary coolant inventory is taken to be 5 GBq t^{-1} Dose Equivalent Iodine (DEI) rather than 16 GBq t^{-1} DEI.

842. The RP has provided a commentary to justify the updated chemistry assumptions, which compares the results for effective doses should the assumption regarding iodine speciation be implemented (Ref. 184). However, no overall comparison of the combined effect of the chemistry assumptions is made, nor is a sensitivity analysis provided, which might provide useful insight into the proposed changes. The RP has provided arguments for each of the amended chemistry assumptions, in an attempt to justify that they remain conservative. Whilst individual assumptions may be conservative, no attempt has been made by the RP to assess the overall impact that all of the changed assumptions will have, and whether the proposals taken together are conservative. Whilst I am content that this is a reasonable position to reach at the end of GDA, the licensee should provide further analysis to consider the impact that these assumptions have, and I consider this to be an Assessment Finding.

AF-UKHPR1000-0164: The licensee shall demonstrate that the chemistry assumptions used in the steam generator tube rupture analysis are conservative, and provide a sensitivity analysis to demonstrate the significance of the chemistry assumptions.

843. In the revised analysis, the RP also indicated that a new operating control would be placed on the primary circuit coolant activity of 5 GBq t^{-1}, in addition to the limit of 16 GBq t^{-1}; the new operating control of 5 GBq t^{-1} has been used in the new analysis, which has been assessed by Fault Studies colleagues (Ref. 117). The revised assumptions regarding iodine speciation and this proposed lower limit both reduce the radiological impact of the fault by a broadly similar amount, around a factor of three.

844. I sampled the updated documents to ensure that these new measures had been adequately described in the generic safety case. The wording in the updated chemistry documents implies that the lower limit is only relevant for the SGTR analysis and does not require the operator to take any action (Ref. 30, Ref. 31). Although the RP indicated that operator actions associated with the new limit of 5 GBq t^{-1} would be developed during the site-specific stages, I consider this to be an important limit, and due to a lack of description of the associated actions during GDA, I consider this to be an Assessment Finding.

AF-UKHPR1000-0165: The licensee shall ensure that the primary circuit activity limit of 5 GBq t^{-1} assumed during Generic Design Assessment is underpinned within the safety case, and shall justify any associated plant or operator actions.

845. My TSC made a brief comparison of the offsite releases reported by the RP to check for consistency between the present and the former revision (Ref. 178). The comparison found that all of the offsite releases are a factor of 3.2 lower in the latest revision for all radionuclides in the first two hours, consistent with the reduction in coolant activity level from 16 GBq t^{-1} DEI to 5 GBq t^{-1} DEI.

846. In the two-eight hour timeframe this relationship is maintained for all the nuclides except for the iodine species, which are lower by a factor of around 4.6. The RP explained that differences in the assumptions about iodine release in the new method (as described above), gave results that were not proportional to the reduction in coolant activity, (Ref. 3).

4.5.3.2 Strengths

847. The safety case analysis provided by the RP was coherent and is generally consistent with RGP. Whilst the underlying chemical processes have not been described in detail, the RP has demonstrated that the analysis is conservative, and the simplified best practice methodology has been justified.

4.5.3.3 Outcomes

848. Based upon the assessment of the DBCs Source Terms Analysis in UK HPR1000 described in sub-section 4.5.2 above, I have identified two Assessment Findings which need to be addressed by the licensee. They concern the chemistry assumptions used in the steam generator tube rupture analysis (AF-UKHPR1000-0164) and the underpinning of the primary circuit activity limit within the wider safety case (AF-UKHPR1000-0165).

4.5.3.4 Conclusion

849. The RP has provided an adequate justification of the SGTR source term. Following the change of approach made to the chemistry assumptions in the accident scenario, the RP has sought to justify that these remain conservative. Whilst a reasonable position

has been achieved for the end of GDA, further work will be required to substantiate these assumptions.

850. Based on the outcome of my assessment of the Design Basis Source Terms Analysis in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and that the RP has made an adequate case to support GDA.

851. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used the guidance contained in SAPs FA.1, which required design basis analysis to be performed to demonstrate that risks are reduced SFAIRP. I also used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors and chemistry assessment and sets out guidance on the assessment of chemistry assumptions in accident scenarios, and NS-TAST-GD-006 which concerns Design Basis Analysis (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the safety case.

4.5.4 Chemistry Assumptions in Design Extension Condition Analysis

852. I reviewed two main documents provided by the RP regarding DECs, for the UK HPR1000 (Ref. 185, Ref. 186). The RP divides DEC faults into sub-categories; DEC-A are for accidents where the main line of protection fails, and DEC-B are for accidents which result in core melt. The chemistry-related phenomena that are expected to be considered in a DEC assessment are as follows (Ref. 162):

- core degradation and relocation;
- In-Vessel Retention (IVR);
- combustible gas production and control;
- fission product release from the core;
- fission product transport through the RCP [RCS] and release to containment;
- fission product behaviour in containment (mainly iodine chemistry); and
- radionuclide control in containment (sprays, pH control, filtration).

853. The UK HPR1000 submissions encompass all of these main areas. The ‘Accident Chemistry Methodology’ report identifies the chemistry aspects (fission product control, combustible gases, corium melt) considered in the assessment, and gives a high-level description of the key phenomena and evaluation methods (Ref. 174). Assessment of each of these aspects is described in the following sections.

4.5.4.1 Radionuclide Release from Design Extension Condition Accidents

854. The radioactive releases have been calculated by the RP using the ASTEC code developed by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) (originally in conjunction with Gesellschaft für Anlagen und Reaktorsicherheit). The ‘Severe Accident Source Terms Analysis’ report describes the relevant parts of the model and gives results for two representative initiating sequences: LB-LOCA and LOOP (Ref. 185). LB-LOCA was chosen as it represents the fastest release of coolant and fission products from the primary system. LOOP was chosen as it is associated with an unfiltered ground-level release to the environment. Additional details of the ASTEC models and their validation are given in the ‘Applicability Assessment’ and the ‘ASTEC Validation Report’ (Ref. 187, Ref. 188). The ‘Sensitivity Studies on Key Parameters of Severe Accident Source Terms Analysis’ report, shows the effect of changing selected modelling assumptions with the LB-LOCA scenario on the calculation results (Ref. 186).

855. Ultimately, the quantity and form of radioactivity released to the environment are determined by the behaviour of the fission products in the containment, and this can be affected by chemistry, particularly in the case of iodine. However, the amounts of fission products released from the fuel and transported through the RCP [RCS] clearly have an important influence on potential releases, as well as defining the heat and radiation sources that determine the conditions within the containment. I therefore decided to sample several of the DEC scenarios, to ensure that the RP has taken account of all of these aspects.

Fission Product Release from the Core

856. Twenty-five elements are considered in the source term calculation, based on the element groupings in NUREG-1465 (Ref. 179); these include all of the top 20 fission products by activity at shutdown listed in relevant guidance (Ref. 189). The inventories are calculated by the PALM code, taking account of the fuel composition and operating parameters of the UK HPR1000 (Ref. 190). The initial mass inventories are the end of cycle values and the initial nuclide activities are the maxima over the reactor lifetime.

857. The fission product release fractions from the core are calculated by the ELSA module in ASTEC. This uses a semi-empirical approach in which the fission products are classified as volatile, semi-volatile or low-volatile, and the release kinetics for each class are calculated assuming a single limiting mechanism: diffusion in solid fuel grains, evaporation/mass transfer, and uranium volatilisation. This means, for example, that the same release kinetics is assumed for all the volatiles (for example xenon, iodine, caesium and tellurium), though the release of tellurium may be modified by interaction with the Zircaloy cladding. The release models have been validated against a range of experimental data, and I am content that it is a reasonable approach (Ref. 187, Ref. 164).

858. Release of fission products from a molten pool is calculated assuming mass-transfer limited evaporation from the free surface, assuming chemical equilibrium in a well-mixed pool. The experimental database is more limited for this phase, and although it is not clear from the documentation how the code treats release of fission products dissolved in the oxide phase through an overlying metal phase, low releases during the late phase would be expected for several reasons:

- most of the fission products will be located in the oxide layer, and cannot penetrate the overlying metal layer;
- there is no steam or hydrogen flow out of the lower head to drive the release of any fission product vapours; and
- any low-volatile fission products released from the molten pool would be extensively retained in the cooler structures above.

859. For the volatile fission products (such as caesium, xenon, iodine and tellurium) an initial gap release occurs when the cladding fails; these gap inventories are defined by the RP, and the results show that the values for caesium, xenon, iodine and tellurium are approximately 4%, 2%, 1.2% and 0.02% respectively (Ref. 185). All of the volatile fission products are completely released within around one hour after cladding failure, so these values do not have a significant effect on the outcome. The semi-volatiles begin to be released when the core outlet temperature exceeds 650 °C and fuel melting starts; they have release fractions of between five and 40% after around one hour. Around 0.01-0.1% of the non-volatile elements are released at this time, which corresponds to the start of core relocation into the lower head. As expected, practically all of the releases occur before the core relocates into the lower head. I am content that this approach is reasonable, which is supported by the advice provided by my TSC (Ref. 164).

Fission Product Transport in the Reactor Coolant System

860. The SOPHAEROS module of ASTEC treats a wide range of fission product transport phenomena in the RCP [RCS]. The fission product vapour composition in the RCP [RCS] is calculated assuming thermodynamic equilibrium, using data from the Material Data Bank Materials Properties Database, which is a database developed and maintained by IRSN and taking account of the results of various international projects. The SOPHAEROS module has been validated against numerous experimental programmes at different scales and the results are described by the RP as "reasonable" for fission product transport and deposition (Ref. 187). Although there is no consideration of the uncertainties in the model results, the thermochemical quantities for the main fission product species are fairly well established, so this is not likely to be a major source of error as non-equilibrium effects are unlikely to be significant.

861. The release of resuspended material from the RCP [RCS] provides a potential late source of radioactivity in the containment, particularly if changes in flow rate or pressure occur, for example because of RPV failure, reflooding of the core or activation of the EUF [CFES]. The resuspension of fission products was not considered in the RP's sensitivity study, produced to investigate the effect of changing selected modelling assumptions on the calculation results (Ref. 186). The RP stated in the response to RQ-UKHPR1000-1296 that ASTEC includes resuspension modelling, and the sensitivity analysis is only intended to evaluate phenomena having an important impact on the source term (Ref. 3). IRSN (the developers of ASTEC), consider that there are sufficient shortcomings in the ability of DEC tools to model the effect of delayed releases to merit additional experiments in the Organisation for Economic Cooperation and Development (OECD) Experiments on Source Term for Delayed Releases (ESTER) project, and so I asked if this implies that there are significant uncertainties associated with the resuspension modelling (Ref. 191). The RP argues that:

- The EUF [CFES] would only be activated if the containment pressure reached 5.2 bar, by which time any resuspension from the RCP [RCS] would already have occurred.
- An increase of 20-30% in the source term from such a low probability event would have very little effect on the overall radiological consequences.
- As IVR is claimed for DEC mitigation, the effect of pressure changes due to RPV failure or reflooding do not need to be considered.

862. I consider that the calculation of fission product transport through the RCP [RCS], and the amounts and chemical forms of the radioactivity release to the containment, uses appropriate methods. Although the code developer recognises uncertainties in the modelling, particularly with regard to resuspension of deposited radioactivity, the RP's arguments that these would not significantly affect the overall radiological consequences are reasonable.

Fission Product Behaviour in Containment – Model and Assumptions

863. ASTEC treats a wide range of fission product phenomena in the containment, including aerosol agglomeration, deposition, resuspension and wash-down. Nucleation and vapour condensation/evaporation of fission products and structural materials on containment surfaces, which depend on the chemical forms and volatilities of the materials released from the RCP [RCS], are also modelled. There is no treatment of chemical changes in the containment, except for iodine and ruthenium; this is appropriate as such changes are unlikely to occur in the containment and would anyway not affect the physical form (and hence the release behaviour), of the other fission products. Ruthenium chemistry is only potentially important if it is released in significant amounts to the containment; this requires oxidising conditions during fuel

degradation to stabilise the volatile ruthenium oxides. This does not apply to the sequences modelled here so ruthenium chemistry was not considered in the RP's assessment or this assessment.

864. The iodine chemistry model in SOPHAEROS represents the following processes (Ref. 187):

- Reaction in water pools, such as interconversion of the main aqueous iodine forms: I^-, I_2, IO_3^-, HOI by thermal and radiolytic reactions, the reaction of aqueous I_2 with organic materials released from submerged painted surfaces, the decomposition of aqueous methyl iodide (CH_3I) by thermal and radiolytic reactions and reaction of aqueous iodine with submerged steel or silver aerosol surfaces.
- Mass transfer and surface reactions, such as the transfer of species between the atmosphere and water pools, the adsorption/desorption onto/from surfaces, surface reactions, and deposition onto surfaces.
- Reactions in the containment atmosphere, such as the radiolytic oxidation of iodine, and the reaction of gaseous iodine with gaseous organic species.

865. This includes all of the main types of reactions that are expected to occur in the containment, but there is likely to be considerable uncertainty in any predictions of organic iodide formation in ASTEC, due to a lack of reliable experimental data. The modelling of gas-phase radiolysis which converts gaseous iodine species (I_2 and methyl iodide) into solid oxide aerosols is by necessity very simplified, and the reaction mechanisms are not fully understood. Although the ASTEC model is a best estimate of the relevant chemistry phenomena, the RP has not reflected on this uncertainty in the safety case, and how they might impact on the calculated release.

866. The description of the iodine chemistry includes some additional reactions (Ref. 185):

- Gaseous decomposition of caesium iodide and cadmium iodide ($CdI_2(g)$)
- Radiolytic and thermal decomposition of deposited iodine pentoxide (I_2O_5) to I_2

867. In the response to RQ-UKHPR1000-0529, the RP describes how the model for decomposition of deposited iodine oxide aerosols to I_2 was developed from experiments in the recent Source Term Evaluation and Mitigation (STEM-2) programme, which are also cited as the validation tests (Ref. 3). This rather limited experimental basis, and lack of theoretical underpinning, again means that there is considerable uncertainty in this part of the model. Despite an RQ seeking clarification, the RP did not provide any evidence on how much impact these reactions have on the results. A future safety case should include consideration of the uncertainties associated with this aspect of the modelling, and therefore I consider this to be an Assessment Finding.

AF-UKHPR1000-0166 – The licensee shall justify the uncertainties assumed in the analysis of the behaviour of fission products during all relevant Design Extension Condition accidents.

Fission Product Behaviour in Containment – Fission Product Behaviour and Release

868. The 'Severe Accident Source Terms Analysis' describes the evolution of the fission product mass distribution (Ref. 185). Initially most of the fission products are suspended as aerosols, but these are increasingly deposited on surfaces or transferred to liquid volumes. The flooding and overflow of the cavity results in most of the deposited aerosols being washed from surfaces, but until the EHR [CHRS] sprays are activated, the IRWST accounts for less than half the aerosol in liquid volumes. The main effect of the sprays (in addition to reducing the containment pressure), is to

transfer aerosol mass from water in the “bottom” zones of the containment into the IRWST.

869. Caesium is calculated to be released from the containment in a range of chemical forms, including elemental; no chemical changes of fission products other than iodine and ruthenium in the containment are treated by the code, so these are the forms released from the RCP [RCS]. All of the caesium is treated as aerosol in the containment, which is realistic as any gaseous caesium arriving in the containment would react very rapidly with steam (forming caesium hydroxide (CsOH)) and/or other aerosols. I asked my TSC to review the releases of fission products from the inner containment, and to the environment (Ref. 164). My TSC concluded that the results of the RP’s analysis were consistent with their expectations for this type of analysis, based upon their experience of the analysis of fission product behaviour and RGP (Ref. 163, Ref. 164). I am therefore content that the approach taken by the RP is reasonable.

Fission Product Behaviour in Containment – Iodine Chemistry

870. The behaviour of iodine aerosol in the containment is the same as for the non-volatile fission products discussed above. This section focusses on the treatment of chemistry leading to the formation of volatile iodine species, and the importance of these to the overall release of iodine activity. Different aspects of this behaviour were considered in the sensitivity study, which examined the effect of changing various model parameters on the calculation of iodine release (Ref. 186).

871. The amount of iodine calculated to be released from the RCP [RCS] in gaseous form is not documented, although from the response to RQ-UKHPR1000-1297, it can be inferred to be about 0.5% (assuming that the initial iodine pentoxide aerosol peak arises from oxidation of incoming gaseous iodine or I_2) (Ref. 3). In the sensitivity study, the incoming gaseous fraction was varied between 10% and 50%; in all cases the fraction of iodine released from the inner containment in gaseous form was about 1/10th of the fraction entering the RCP [RCS] (Ref. 186). This implies either that gaseous iodine was converted to aerosol, or that it was more effectively removed from the containment atmosphere (or a combination of the two). The response to RQ-UKHPR1000-1297, identifies conversion to aerosol as the dominant mechanism but suggests that this is the result of the relatively low containment temperature (Ref. 3). It is more likely that the conversion would be driven by radiolytic oxidation rather than condensation, but this would not change the overall behaviour.

872. The RP presented evidence which shows that from about one hour after the start of the accident (when most of the release from the RCP [RCS] is complete and aerosols are depositing in the containment), until about 12 hours (when the sprays become operational) most of the aerosol mass is dissolved in water outside the IRWST (Ref. 185). In the response to RQ-UKHPR1000-1297, the RP confirms that iodine chemistry is calculated in the same way for all water volumes in the containment, which are considered to be acidic before the sprays are activated (Ref. 3). However, the RP also clarified that chemistry is not modelled in the reactor pit or in surface water films. These two aspects are considered further in the sections that follow.

873. The main function of the EHR [CHRS] is to remove decay heat from the containment to avoid over-pressurisation leading to containment failure. The EHR [CHRS] consists of two spray trains, each with a dedicated cooling system to remove the heat to the ultimate heat sink, and as stated by the RP in response to RQ-UKHPR1000-0529, the sprays are credited in both containment pressure control and severe accident source term analysis for UK HPR1000 (Ref. 3). As noted above, the EHR [CHRS] has very little effect on the suspended activity because most of the aerosol has already deposited on surfaces or into the IRWST before the sprays are activated. In most cases (except for the noble gases and a few other fission products), more than 70% of

the total activity release occurs in the first 12 hours in the LB-LOCA calculation; for the LOOP calculation this rises to more than 95%. However, use of the sprays would help to ensure that volatile forms are minimised in the longer term.

Fission Product Behaviour in Containment – Volatile Iodine Release from Water Volumes

874. Control of the pH of the water in the IRWST is a key measure to prevent the formation of volatile iodine species, which are key radionuclide in terms of the generation of radioactivity. I therefore sampled the RP's modelling of the water chemistry in the IRWST post-accident.

875. Figure 11 shows the speciation of iodine in the IRWST during a LB-LOCA scenario, as calculated with ASTEC.

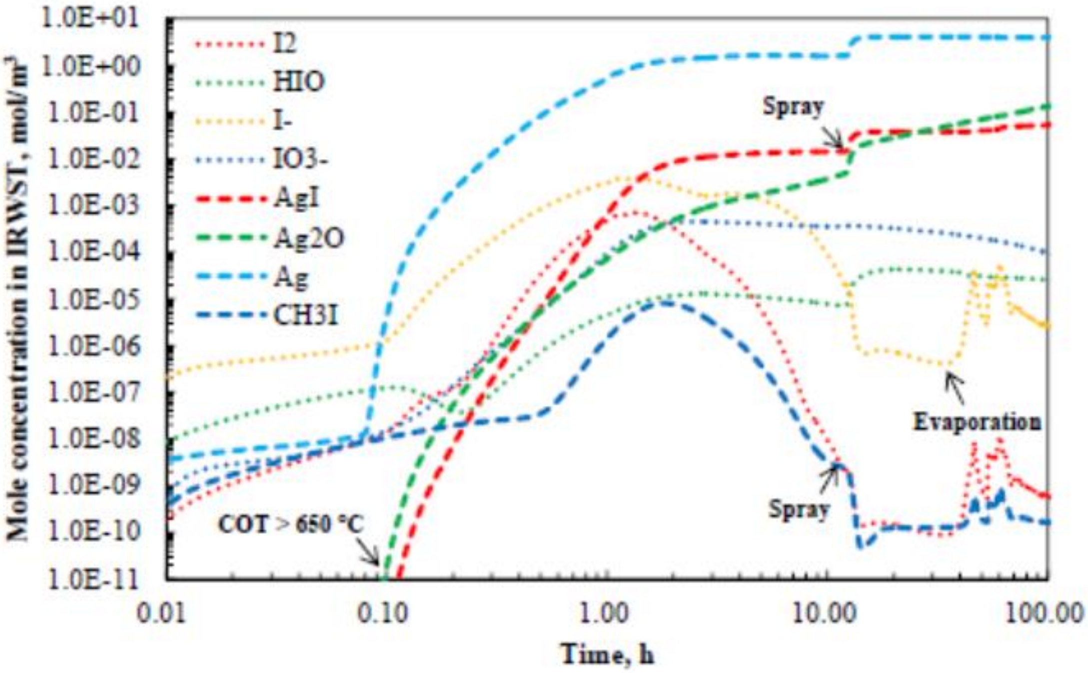

Figure 11: Speciation of Iodine in the IRWST **(Ref. 185)**

876. Most of the iodine in the sump eventually forms silver iodide (AgI), as silver is present in about 100x molar excess (silver is present in this scenario due to it being a component of control rods). The sensitivity study shows that the retention of iodine by silver is most effective during the initial phase where the water pH is low because of the rapid reaction of I_2 with silver metal; at high pH the main reaction is between iodide and oxidised silver, which may limit the extent of the reaction (Ref. 186). As long as the pH of the IRWST is controlled to a high value, release of volatile iodine is very low regardless of the presence of silver.

877. Changes in the IRWST pH in the first eight hours of the accident had no significant effect on the calculated iodine releases. This is unsurprising as the IRWST was in all cases assumed to maintain a pH of 7 throughout most of the accident, and any initial increase in volatile iodine production would have occurred during a period when aerosol releases were dominant. Also, as noted above, a lower pH is favourable to retention of iodine as silver iodide in the IRWST.

878. Although practically all of the iodine is eventually dissolved in the IRWST, this water is injected into the reactor pit to maintain ex-vessel cooling. Based on the figures provided by the RP, at a flow rate of 40 m^3 hr^{-1}, my TSC calculated that all of the water initially in the IRWST, RCP [RCS] etc. would pass through the pit within 62 hours of the

start of active injection (Ref. 185, Ref. 192). The evaporation of water in the pit and recondensation of steam back to the IRWST would lower the concentration of TSP in the IRWST and could compromise its buffering capability. However, iodine and other solutes including nitric acid could also become concentrated in the reactor pit, whose volume is less than 1/10th of the total water in the containment. The pit could therefore become an important source of iodine volatility due to aqueous radiolysis reactions. However, in the response to RQ-UKHPR1000-1297, the RP states that most of the flow out of the reactor pit is liquid water rather than steam, so the concentrations of all the solutes will be similar in the pit and the IRWST (Ref. 3). Although I consider that it is likely that the iodine chemistry in the pit would not significantly influence the calculated releases, the RP has not demonstrated this in the generic safety case, and I consider this to be a minor shortfall.

879. During an accident, the temperature of the water in the IRWST would be expected to increase. Together with pH, the temperature of the water will influence iodine chemistry. However, this has not been considered by the RP. I consider this to be an Assessment Finding.

AF-UKHPR1000-0167 – The licensee shall demonstrate the effect that changes in temperature in the in-containment refuelling water storage tank have on iodine chemistry during severe accidents, and justify the consequences.

880. The RIS [SIS] is designed to maintain the pH of the IRWST above 7 by dissolution of TSP, preventing additional production of volatile iodine from dissolved caesium iodide. For DEC accidents it is important to demonstrate that the amount of TSP in the system is sufficient to counteract the various acid sources. The pH evolution in the IRWST is calculated using a stand-alone module of ASTEC. All of the major sources of acid or alkali production are considered, and the decisions to ignore some of them (carbon dioxide, hydrogen iodide, caesium hydroxide) are justified and reasonable. The results presented by the RP for the DEC accident sequence show that the IRWST pH increases from its initial value of 4.6 to a maximum of 7.7 after about three hours and although this falls slightly over the following ten hours, a pH of well over 7 is maintained for at least 168 hours (Ref. 185).

881. The IRWST pH is determined by the concentrations of boric acid, lithium hydroxide, TSP and nitric acid formed from radiolysis of air in the containment atmosphere. The early increase in the pH is determined by the dissolution rate of TSP into the boric acid solution in the IRWST. Eventually, though, TSP dissolution will be complete but nitric acid will continue to be formed by air radiolysis, so a gradual decrease in pH would be expected. The extent of this decrease will depend on the rate of production of nitric acid, so this is a key factor in the calculation. The nitric acid production rate depends on:

- the G value for nitric acid production in air (the amount of nitric acid produced per unit of energy deposited; and
- the rate of radiation energy deposition in the air (dose rate multiplied by the mass of air).

882. The dose rate used in the pH model is calculated by ASTEC, and this is different from the dose rate calculated by the radiological protection team for equipment qualification and post-accident accessibility, as described in the response to RQ-UKHPR1000-1297 (Ref. 3). The RP has presented data that shows that the dose rate in the containment upper atmosphere remains above 1 Gy s^{-1} until at least 72 hours (Ref. 185); my TSC notes that a rough calculation of the nitric acid production rate suggests that a significant lowering of the pH could be expected at this dose rate (Ref. 163). However, the RP explained in the responses to RQ-UKHPR1000-1297 and RQ-UKHPR1000-1443 that the pH calculation uses the dose rate of the gaseous phase in the IRWST

upper compartment, which is significantly lower (≈ 0.2 Gy s^{-1} at 72 hours), and that it is applied to all of the air in the containment (Ref. 3).

883. The response to RQ-UKHPR1000-1443, shows that if the dose rate in each containment compartment is used to calculate the nitric acid production in that compartment, and all the resulting acid is assumed to be dissolved in the IRWST water, pH control in the IRWST is lost after about 30 hours (Ref. 3). Although this is described as a conservative estimate, no justification is given for the supposed conservatism. It is implied that using the IRWST compartment gas phase dose rate was appropriate because this is in contact with the IRWST water volume; however, nitric acid produced throughout the containment atmosphere would be readily dissolved and transported to the IRWST in the sprays. I therefore consider this to be an Assessment Finding.

AF-UKHPR1000-0168 – The licensee shall demonstrate that the pH control of the in-containment refuelling water storage tank accounts for all sources of acidification.

Fission Product Behaviour in Containment – Volatile Iodine Releases from Surfaces

884. The ‘Severe Accident Source Term Analyses’ show that 10-30 kg of aerosol remains on the surfaces which, depending on the conditions, could be a continuing source of gaseous iodine (Ref. 185). Although this is a small fraction of the total (~1%), it is large compared with the overall iodine release (~0.01%), so a modest decomposition rate could have a significant effect on the airborne iodine concentration and subsequent releases. Fission products deposited on surfaces are considered to remain as aerosols, rather than dissolving in condensate films. Iodine pentoxide is the only iodine aerosol for which decomposition is considered, but this only accounts for a very small amount of the total.

885. In the RP’s sensitivity study, the fraction of the aerosol deposited on the walls that remained after washing was varied between 20% and 100% of a “typical” deposited mass of 700 kg (Ref. 186). It can be inferred that the distribution in Figure 11 corresponds to a residual fraction of about 30% (500 kg of the deposited aerosol is washed off the surfaces as the cavity floods).

886. Although the description of the calculation in the response to RQ-UKHPR1000-1297 suggests that the concentration of aerosol remaining in a water film on the wall is an important consideration, the residual aerosol is treated as deposited rather than dissolved in a surface water film (Ref. 3). Although the RP does not go on to discuss this, the small changes in calculated iodine releases therefore presumably arise from the decomposition of the aerosol deposits, though the lack of correlation between either the total or gaseous iodine release and the residual fraction suggests that this is not the only effect.

887. My TSC calculated that the dissolution of 10% (200/2000 kg) of the iodine inventory in a surface water film of 9 m^3 would give a concentration of about 0.002 mol dm^{-3}. Any volatile iodine formed by radiolytic reactions within this film would be more readily released to the atmosphere than from bulk water volumes because of the very high surface area/volume ratio. Although I have identified areas where the RP has not considered the full impact of results of their analysis, (such as the formation of volatile iodine from surface water films, and the decomposition of deposited aerosols), the amount of deposited iodine is small compared with that in airborne aerosol and other unbuffered water volumes, such as condensate films, during the 12 hours before the sprays become operational. It is reasonable to assume, therefore, that these mechanisms would not significantly increase the iodine release during this period. In both of the accident scenarios described by the RP (LB-LOCA and LOOP), the containment sprays are always used. Once the sprays are operating, most of the

iodine should be washed off the surfaces and retained in the IRWST, as long as a sufficiently high pH is maintained, which is discussed further in subsequent sections of this report. However, the RP has not demonstrated that the two scenarios considered are the most penalising for iodine production; certain scenarios that do not involve the use of containment sprays could produce greater volumes of iodine. I therefore consider this to be an Assessment Finding.

AF-UKHPR1000-0169 – The licensee shall demonstrate that the source term associated with iodine production during severe accidents has considered all relevant scenarios, to demonstrate that the risks associated with iodine production have been reduced so far as is reasonably practicable.

888. Later on in the LB-LOCA scenario, caesium and iodine are resuspended from the primary circuit, and are released into the containment. The RP provides evidence to show that the amount of iodine deposited on surfaces increases by an order of magnitude, and accounts for about 10% of the total iodine in containment through the remainder of the calculation. During this time, iodine is released from surfaces, such as the decomposition of aerosol deposits, or volatile organic iodide formation from reactions with paint or other organic pollutants. The results from ASTEC provided by the RP show that the gaseous iodine concentration is too low to make a significant contribution to the overall activity release.

889. Despite the findings that I identified in my assessment of the release of volatile iodine from surfaces, I am content that the overall approach is reasonable.

Fission Product Behaviour in Containment – Overall Volatility and Iodine Releases

890. The RP reported in the response to RQ-UKHPR1000-1297 that the mass of iodine remaining in gaseous form is extremely low, accounting for only about 10^{-5} of the total in the LB-LOCA calculation (Ref. 3). My TSC estimated this to equate to an overall Iodine Partition Coefficient (IPC, iodine concentration in water/gaseous iodine concentration, in volumetric units) of 3×10^{6}. My TSC considered that this coefficient was high, despite the fact that the production of gaseous iodine from the IRWST would be suppressed by effective pH control. Considering the results from the latest PHEBUS research, my TSC concluded that the radiolysis model in ASTEC may therefore somewhat underpredict iodine volatility from irradiated water at high pH, though the conclusion that it is very low is valid (Ref. 193, Ref. 164).

891. The recirculation of IRWST water containing dissolved iodine through the spray system would be expected to maintain a constant concentration of gaseous/airborne inorganic iodine similar to that given by this measured IPC. Given the uncertainties in elements of the modelling, the possibility that the gaseous iodine concentration could be an order of magnitude higher than that given by the calculations cannot be ruled out. Even if this were the case, though, the airborne iodine mass would only be increased by about 10-20%, and the release would still be dominated by aerosol (although doses would still be dominated by other species).

892. The RP states that iodine is calculated to be released from the containment very predominantly as aerosol, with only 0.07% of the total release in obviously gaseous form (I_2, iodine and methyl iodide) (Ref. 185). The fraction is much lower than the reference values in NUREG-1465 (5% gaseous of which 0.15% organic) (Ref. 179). The latter values have been used to calculate the speciation of the activity released to the containment, which affects the filtration efficiencies assumed in the source term calculations. This should lead to overprediction of the iodine activity release as long as the pH is controlled within the IRWST. This is a reasonable approach in view of the remaining uncertainties in the volatile iodine production mechanisms. However, as discussed above, it may not be conservative some time after commencement of the

accident, when the aerosol release rate is very low and the gaseous fraction may be higher than 5%. This would not significantly change the overall release, which is dominated by aerosol.

893. In conclusion, I consider that the safety case satisfactorily demonstrates that the contribution of gaseous forms to the overall release of iodine will be minimal as long as the mitigation measures (pH control, containment sprays) operate as expected. Although there are uncertainties in the models used, these are very unlikely to be large enough to threaten this conclusion. I am content that the RP has adequately considered fission product behaviour, and that any uncertainties in the analysis are bounded by the conservatisms that I have described.

4.5.4.2 Core Degradation and In-Vessel Retention

894. The IVR strategy report, describes the strategy, design and analysis of the IVR system (Ref. 194). The strategy is chosen by the RP to deal with a core melt accident, and uses the reactor pit flooding system, which is part of the EHR [CHRS]. After a postulated DEC-B accident, water is injected into the reactor pit and submerges the RPV in the case that core degradation and melting occurs (the molten core material is known as the corium). The cooling water can achieve natural circulation in the reactor pit, through which decay heat can be removed from the corium by external cooling. Thereby, containment of the core is maintained within the RPV. A diagram of this system is shown in Figure 12.

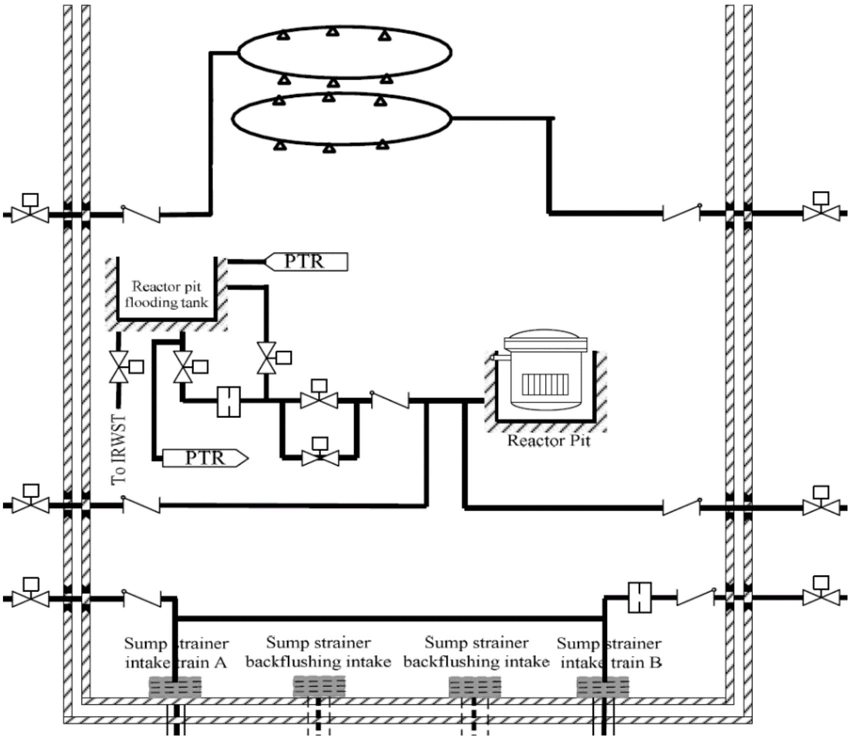

Figure 12 Part of the EHR [CHRS] System, Showing the IVR System

895. Many aspects of core degradation and IVR are beyond the scope of a Chemistry assessment, and have been assessed by my Severe Accident Analysis colleagues (Ref. 167). However, several aspects of the core melt and IVR analysis depend on how the chemistry has been treated, and these form the basis of my assessment in the proceeding paragraphs.

896. The current review focusses mainly on the ASTEC calculations used by the RP to predict the progression of an accident through core degradation to the formation of a molten pool in the lower plenum, and how the results are applied in assessing whether

the RPV will maintain its integrity. ASTEC is used by the RP to produce deterministic analysis to determine the mass and composition of the molten pool. A key consideration in these calculations is the structure of the melt pool that forms in the lower head, and in particular the thickness of the overlying metallic layer. Since this is a major pathway for heat ejection from the melt to the RPV wall, a thinner metal layer will result in a higher local heat flux for a given decay power in the melt. This is referred to in the following discussion as the “focussing effect”.

ASTEC Modelling – Core Degradation and Relocation

897. ASTEC includes several modules that are involved in the calculation of core degradation and relocation:

- ICARE (core degradation, relocation and phase separation)
- CESAR (heat transfer to external cooling water)
- ELSA (fission product release)

898. The ICARE model takes account of a wide range of thermal, mechanical and chemical interactions to calculate the physical changes that occur during the degradation and relation of the core. The main chemical processes are:

- Steam oxidation of zirconium and stainless steel, resulting in heat generation and hydrogen production.
- Solid-solid and solid-liquid interactions (uranium dioxide dissolution by zirconium, zirconium dissolution by liquid silver-indium-cadmium alloy, dissolution of zirconium in solid stainless steel, oxidation/dissolution of uranium-oxygen-zirconium magma), that determine the physico-chemical forms and properties of the core materials.

899. Both oxidation reactions have been validated experimentally, using data drawn from publicly available databases (Ref. 164). In the context of IVR, the fission product release is important as it determines the residual decay heat in the lower head. There is an extensive validation database for fission product release from fuel rods and debris beds. Although the models for releases from molten pools are less well validated, the RP calculates these to be negligible, so I am content that any uncertainties are not important in the IVR context.

900. My assessment of the modelling of fission product release by the ELSA module can be found in sub-section 4.5.3.1.

ASTEC Modelling – Corium Behaviour in the Lower Plenum

901. The ICARE model accounts for (Ref. 187):

- the stratification of corium in the lower head into up to three liquid and two debris layers;
- heat transfer between the layers and with the vessel wall;
- radiative heat transfer from the upper layer to the core support plate;
- oxidation of the corium pool; and
- rupture of the vessel lower head by melt-through or mechanical failure.

902. The main impact of chemistry is in the stratification modelling, which determines the amount (thickness), composition and thermophysical properties of the different melt layers and the distribution of fission products between them. ASTEC has two options: the “simple phase separation” (SPS) and “thermochemical equilibrium phase separation” (TEPS) models. These aspects are discussed in the following sections.

ASTEC Modelling – Phase Separation Modelling

903. Although the TEPS is the default model, the SPS was used in the calculations presented in the Strategy report, and as the base case in the sensitivity studies (Ref. 195, Ref. 194). The RP clarified in the response to RQ-UKHPR1000-1168 that in the SPS model the corium forms two layers, predominantly constituted of oxides or metals, based on material density (Ref. 3). The TEPS model first predicts the pool composition based on the miscibility gap in the uranium-oxygen-zirconium-iron quaternary system. The local thermochemical equilibrium is then considered at the interface(s) between the layers, which might result in changes in the composition and arrangement of the layers. The equilibrium compositions are established using the NUCLEA database, (developed by IRSN alongside ASTEC), which is based on tests with prototypical materials, including the MAterial SCAling (MASCA) programme, (which was developed by the OECD). As well as the main metallic and oxidic components, the database includes representative fission products (barium, strontium, lanthanum and ruthenium) in both metallic and oxide forms (except for ruthenium, which forms volatile oxides) (Ref. 196). The kinetics of stratification in the TEPS are influenced by the chemical interactions between metal and oxide crusts. The RP acknowledged that the modelling is still under development and there are significant uncertainties in some of the variables.

904. The POOLMODE model, (a module of ASTEC), calculates the thickness and thermal resistance of crusts that form around the corium pool. The RP clarified in the response to RQ-UKHPR1000-1319 that the crust is not physically represented: it has no physico-chemical properties and holds no residual power (Ref. 3). It is considered in the heat transfer calculation as a thermal resistance defined by the mass and energy balance. When the corium is above its solidus temperature (defined as the temperature at which melting begins, below which there is no thermal crust resistance), the crust thermal resistance depends on the properties of the corium layer (liquid fraction, volume, surface area, temperature, melting point, conductivity), and vessel wall (temperature, conductivity). The effect of the crust will therefore depend on the composition and thermochemistry of the melt.

905. In the response to RQ-UKHPR1000-1168, the RP described the evolution of the melt in TEPS (Ref. 3). Formation of a heavy metal layer below the oxide is predicted initially, with a lighter (steel) metal layer above. Then as uranium, zirconium, iron and oxygen diffuse between the layers to maintain thermochemical equilibrium at the interfaces, the heavy metal will start to be oxidised when the oxide layer reaches a certain degree of oxidation. The transfer of oxygen into the metal layer leads to a reverse transfer of uranium and zirconium into the oxide layer; when the metal layer becomes less dense than the oxide, inversion of stratification will occur. The metal layer will be significantly superheated, probably leading to a temporary high heat flux to the RPV wall when it reaches the top of the pool. In the response to RQ-UKHPR1000-1319, the RP clarified that the three-layer configuration, with the heavy metal below the oxide, is transient and the steady-state configuration only has two layers (Ref. 3).

906. My TSC reviewed the different steady state melt compositions which are produced by the two models (TEPS and SPS) (Ref. 164). As the TEPS model takes greater account of the chemical properties of the corium components, allowing for a more nuanced description of the composition of the pool and the distribution of fission products within it, it should ideally give a more realistic representation of the steady state thermal behaviour than the SPS model. However, as noted in the response to RQ-UKHPR1000-1319, whilst the model parameters have been validated by some experiments, the analysis of the inherent uncertainties in the model is still ongoing by both developer and users (Ref. 3). There is no international consensus on whether the three-layer model reflects reality, and the licensee should continue to follow the ongoing research in this area. I consider this to be a minor shortfall.

ASTEC Sensitivity Studies

907. The RP provided a large number of sensitivity studies to support the use of ASTEC in IVR analysis, although the majority are not relevant for this assessment. I note that the parameters to be studied in the sensitivity calculations were defined from a Phenomena Identification Ranking Table (PIRT), recently developed as part of a Horizon-2020 (H2020) European project in IVR, which takes account of the views from experts and developers of all the main severe accident codes and others that are specific to IVR (Ref. 169). The use of such recent information is a strength of this aspect of the case.

908. The RP identified several phenomena that were selected for the sensitivity analysis, based upon the PIRT, which included chemical interactions (zirconium and steel oxidation correlations) and stratification kinetics (phase separation modelling: SPS/TEPS). I sampled the PIRT to provide confidence that the most significant phenomena had been selected for analysis by the RP. Although there are some small differences in approach, given the constraints of the available tools, I am content that the sensitivity studies method is adequate. The failure criterion is not met for any of the sensitivity studies, either through rupture of the RPV, or if the local heat flux is greater than the Critical Heat Flux (CHF). No cliff edge effects were identified either.

909. I raised RQ-UKHPR1000-1257 to query some aspects of the modelling, and in response the RP gave details of the only calculation where RPV failure occurred (Ref. 3). To investigate the impact of volatile fission product release, the fraction of the total decay heat assigned to uranium dioxide was varied. With 80-90% of the decay heat remaining in the uranium dioxide, the RPV was not breached. However, with 100% of the decay heat remaining in the uranium dioxide, the RPV failed. The fact that RPV failure is only predicted when this very unrealistic assumption is made supports the RP's contention that there is no cliff edge effect.

Safety Functional Requirements of In-Vessel Retention

910. The safety functional requirements of the IVR system are mainly concerned with achieving adequate flow rates, and the control and monitoring of flow rates and water levels (Ref. 192). The only chemistry considerations are in the water quality requirements. Boric acid will be present in the injection water, both in the Reactor Pit Flooding Tank (RPFT), and the IRWST. The RP clarified that the concentration of boric acid in the RPFT is the same as that in the IRWST (1300-1400 mg kg^{-1}), to avoid dilution of the IRWST water, either inadvertently (as the RPFT is located above the IRWST) or when water is transferred to the IRWST during refuelling.

911. As water evaporates from the reactor pit, the concentrations of boric acid and TSP will increase in the pit and decrease in the IRWST. However, as discussed previously (see sub-section 4.5.4.1), this is not expected to lead to significant differences in the concentrations of solutes in the pit and the IRWST. The RP clarified in the response to RQ-UKHPR1000-1319 that the concentration of boric acid in the pit increases from 1300-1400 mg kg^{-1} to about 3500 mg kg^{-1} during the ten hour passive injection phase (Ref. 3), conditions which are not expected to affect RPV integrity. In the response to RQ-UKHPR1000-1168, the RP also cited experimental studies showing that both boric acid and TSP would have a beneficial effect on the heat removal capacity of the external vessel cooling system compared with deionised water which was used in the REVECT experiments used to determine the CHF profile of the lower head (Ref. 3).

4.5.4.3 Strengths

912. The RP has conducted a detailed analysis of the chemistry of radionuclide releases during DEC accidents, which although have potential uncertainties, should give more reliable results than some alternative methods.

913. The RP has used appropriate methods to analyse the chemistry assumptions involved in the IVR safety case, and I am content that it is adequate for the UK HPR1000.

914. The RP has used the latest information and recommendations from expert groups in the development of the IVR sensitivity studies.

4.5.4.4 Outcomes

915. Based upon the assessment of the chemistry aspects of DEC analysis in UK HPR1000 described in sub-section 4.5.4 above, I have identified several Assessment Findings which need to be addressed by the licensee. These concern the justification of the uncertainties assumed in the analysis of the behaviour of fission products (AF-UKHPR1000-0166), the demonstration of the effect that changes in temperature in the in-containment refuelling water storage tank have on iodine chemistry during severe accidents (AF-UKHPR1000-0167), the demonstration that the pH control of the in-containment refuelling water storage tank accounts for all sources of acidification (AF-UKHPR1000-0168) and the demonstration that the source term associated with iodine production during severe accidents has considered all relevant scenarios, to demonstrate that the risks associated with iodine production have been reduced SFAIRP (AF-UKHPR1000-0169).

916. I also identified two minor shortfalls in this part of my assessment.

4.5.4.5 Conclusions

917. The RP has adequately described the chemistry effects during DEC accidents, and justified that the overall chemistry assumptions are conservative.

918. Based on the outcome of my assessment of the chemistry in DEC accident analysis in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

919. I have identified four Assessment Findings as a result of this part of my assessment.

920. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used the guidance contained in SAPs FA.1, which requires DEC analysis to be performed to demonstrate that risks are reduced SFAIRP. I used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors and chemistry assessment and give guidance on chemistry assumptions used in accident analysis, and NS-TAST-GD-007, which concerns Severe Accident Analysis (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the generic safety case.

4.5.5 Hydrogen Generation and Control

921. During a number of design basis and potential DEC sequences, the possibility exists for the generation of hydrogen-rich atmospheres within the containment of any LWR. The major concerns regarding hydrogen are that the pressure or temperature loads from combustion may damage the containment or that important safety-related equipment may be damaged. In order to assess the possible threats, it is necessary to understand how hydrogen is produced, and how it is mitigated by the generic design of the UK HPR1000.

922. There are several production routes for hydrogen in the UK HPR1000:

- steam oxidation of Zircaloy and stainless steel during core degradation;

- Molten Core-Concrete Interactions (MCCI), which also produce carbon monoxide (CO);
- corrosion of metals by water in the containment;
- radiolysis of water in the RCP [RCS] or containment; and
- release of hydrogen dissolved in the primary coolant.

923. The distribution and combustion of hydrogen in the containment are determined by physical processes which are assessed elsewhere (Ref. 167). My assessment has focussed on the assumptions made by the RP about hydrogen generation, and the mitigation measures that are included within the generic design of the UK HPR1000. The RP has made a number of arguments and sub-arguments for this topic, which seek to justify that the hydrogen concentration in the containment under fault/accident conditions is controlled (Ref. 5).

4.5.5.1 Hydrogen Mitigation

924. Hydrogen is mitigated in UK HPR1000 by 29 Passive Autocatalytic Recombiners (PARs) (11 small and 18 large), of which two are qualified to a higher-level safety classification (F-SC2) for DBCs. All 29 are credited in the DEC analysis. The removal rate of hydrogen by PARs is the Siemens correlation used in ASTEC, simplified to remove the effect of carbon monoxide (Ref. 187). I raised RQ-UKHPR1000-1296 to obtain more information about the hydrogen mitigation philosophy of the UK HPR1000 (Ref. 3). The response to RQ-UKHPR1000-1296 compares the values of the coefficients in ASTEC with those determined experimentally by the supplier of the PARs used in FCG3, which are assumed to be the same for UK HPR1000. However, I expect a correlation to be provided between the PAR included within the UK HPR1000 and experimental data to demonstrate that the claimed operation is supported by evidence. This will depend on the detailed design, which will be produced following GDA. I have gained sufficient confidence for the proposed PAR design for GDA, but I consider the lack of a comparison with experimental data to be a minor shortfall. An assessment of the sensitivity studies that the RP performed on the functionality of the PARs is described in sub-section 4.5.5.1.

4.5.5.2 Design Extension Conditions

Containment Combustible Gas Control System

925. The RP’s combustible gas control assessment report describes the design basis for the EUH [CCGCS], the chemistry aspects (relating mainly to the production and mitigation) of combustible gases, and the results of ASTEC calculations for selected DEC scenarios (Ref. 197). The basis is:

- The average hydrogen concentration in containment during and following a DEC accident is maintained below 10 vol%.
- If local flame acceleration occurs, the integrity of the containment is maintained.

926. The loads from flame acceleration are assessed separately by CFD modelling. For UK HPR1000, only high-temperature oxidation reactions are considered in the DEC assessment; MCCI is precluded by IVR. The amount of hydrogen present in the primary coolant is trivial compared with the other potential sources, as shown in the DBC assessment (see sub-section 4.5.2). Corrosion and radiolysis are slow, long-term mechanisms which would produce hydrogen at a slower rate than it could be removed by PARs. Although the RP provides a reasonable justification for the assumption about corrosion, no justification was presented for radiolysis, which could be significant in a DEC accident (Ref. 197). The response to RQ-UKHPR1000-1296 estimates the hydrogen generation from radiolysis in the IRWST over seven days, using conservative assumptions (Ref. 3). The maximum production rate is two orders of magnitude lower than the capacity of the 29 PARs credited for a DEC accident. The total mass of

hydrogen produced in seven days is 180 kg, which is less than half that calculated to be generated by core degradation over a much shorter timescale. I am therefore content that these assumptions are conservative.

Assessment Procedure and Conclusions

927. ASTEC is used to calculate the hydrogen production and distribution, and an initial assessment of the flammability of the predicted compositions is made using Shapiro diagrams (which show deflagration and detonation regions for hydrogen/steam/air mixtures). The more detailed CFD analysis using the GASFLOW code uses the mass and energy releases calculated by ASTEC to evaluate the gas distribution and PAR performance. The relevant chemistry models in ASTEC are the correlations for Zircaloy and stainless steel oxidation in steam, and the PAR efficiency correlation.

928. The initial selection of the DEC scenarios used in the assessment was based on sensitivity calculations done for the FCG3 HPR1000. These were used to down-select seven representative and limiting scenarios for the UK HPR1000 study. The results of ASTEC calculations for these seven cases are discussed in the sensitivity study report, mainly in terms of the total amount of hydrogen produced, the maximum production rate, and the peak local hydrogen concentration (Ref. 198). In general, the sequences with high production rates give lower total hydrogen releases because the accident proceeds more quickly so relocation occurs earlier. The maximum local concentration was above 10 vol% for the three PZR break scenarios. The maximum Adiabatic Isochoric Complete Combustion (AICC) pressure was below the containment design pressure (0.52 MPa) in all cases, but this is mainly determined by the amount of steam in the containment (Ref. 20).

929. A more detailed assessment of the EUH [CCGCS] effectiveness was made for three scenarios:

- Case 2: 5.0 cm SB-LOCA at the top of the PRZ, which gave a high total hydrogen production and a concentration of 10.5 vol% in the PRZ compartment.
- Case 4: Station blackout, with delayed depressurisation, which gave the highest total hydrogen production and AICC pressure.
- Case 7: 7.5 cm Intermediate Break Loss of Coolant Accident (IB-LOCA) at the top of the PRZ with delayed depressurisation, which gave a high hydrogen production rate and the highest local concentration (13.3 vol% in the PRZ compartment).

930. Case 7 was also used for the ASTEC model sensitivity studies, discussed below (Ref. 198). In each case, the hydrogen concentration in the affected compartment(s) and the containment dome were plotted for calculations with and without PARs activated. As expected, disabling the PARs significantly increased the calculated hydrogen concentrations some time after commencement of the accident, though the peak values calculated during the degradation phase were not greatly affected. In Cases 2 and 7, the peak hydrogen concentrations in the PZR were within the deflagration region of the Shapiro diagram. In all cases, the long term hydrogen concentrations with PARs operating were well below the flammability limit of 4 vol%.

931. The assessment concludes that the global hydrogen risk is mitigated by the EUH [CCGCS]. The two cases where the local hydrogen concentration exceeded 10 vol%, even with the PARs operating, are judged to present a possible hydrogen risk. This has been analysed further by CFD calculations (see ONR’s Severe Accident Analysis Step 4 assessment report for further details (Ref. 167)).

Sensitivity Studies on Hydrogen Risk

932. The sensitivity of the hydrogen risk to uncertainties in the assessment is determined by the modelling of:

- the amount and timing of hydrogen release;
- hydrogen distribution within the containment; and
- hydrogen removal by the EUH [CCGCS].

933. The RP's sensitivity study describes ASTEC calculations to evaluate the impact of varying parameters and model assumptions relating to the amount and timing of the hydrogen release, and the removal by EUH [CCGCS] on the results and conclusions described in the assessment report (Ref. 198, Ref. 197). The effect of hydrogen distribution within the containment is considered with respect to the different break locations in the accident scenarios and in the CFD calculations; there are no chemistry aspects to these analyses, so they are not discussed further here.

934. Based on the scenario analysis described in the previous section, Case 7 (IB-LOCA at the top of the PRZ) was selected as the bounding case for the sensitivity study. This gave the highest local peak hydrogen concentration (13.3 vol%), so potentially the highest flame acceleration risk, though the total hydrogen production was amongst the lowest, presenting less challenge to the PARs compared with other sequences. However, the effect of PAR operation on the long-term hydrogen concentrations was similar in all the cases studied, so the modelling sensitivities would be expected to be similar (Ref. 197).

935. The sensitivity study looked at the effect of:

- the core degradation modelling, which determines the amount of hydrogen released;
- the timing of containment spray operation, which influences the composition of the atmosphere; and
- the performance of the PARs, which determines the removal rate of hydrogen from the containment atmosphere.

936. As discussed previously (sub-section 4.5.5.2), the high-temperature oxidation of metals by steam during core degradation is the main hydrogen production mechanism in a DEC accident. The sensitivity study considers uncertainties in the zirconium-steam reaction model and the effect of changing the fuel melting temperature, which determines the duration of the oxidation reaction, from the default value of 2500 K. It is reasonable to focus on the dominant zirconium oxidation reaction, as steel oxidation only contributes 10-15% of the total hydrogen; however, the approach taken is simply to compare the predictions of the alternative oxidation models available in the ASTEC code, which do not necessarily reflect the uncertainties in the underlying experimental data.

937. The effect of the different oxidation models on the hydrogen production is fairly small, with maximum generation rates of 0.6–0.65 kg s^{-1} and total released masses of 475-501 kg. Changing the fuel melting temperature produced a wider range of results, with maximum generation rates up to 0.73 kg s^{-1} and total released masses of 445-621 kg. It is noted that the CFD analysis used a hydrogen mass of 900 kg, equivalent to 100% zirconium oxidation, and scaled-up production rates with a maximum of about 1.1 kg s^{-1}, so the hydrogen risk from these calculations clearly bounds any uncertainties in the ASTEC calculations.

938. Activating the containment sprays decreases the steam concentration, potentially shifting the composition of the atmosphere into the deflagration or detonation regions by increasing the hydrogen and air fractions. With spray activation before the start of

hydrogen generation, or at the time of peak hydrogen concentration in the PRZ compartment, the calculated peak hydrogen concentrations in the containment dome are increased above 4 vol%, though without exceeding 10%. There is a slight increase in the peak hydrogen concentration in the pressuriser compartment (from 13.3 to 13.6 vol%) moving the local composition further into the deflagration zone. The hydrogen level is brought down to the same level as the base case at around ten hours by the action of the PARs, and increases only slightly when the sprays are activated 12 hours after the start. The scenario used for the sensitivity analysis was chosen as it gave the highest local peak hydrogen concentration. However, this does not mean that it is necessarily the most penalising for this particular scenario. Although the results from the sensitivity analysis are reassuring, the RP has not adequately demonstrated that the chosen scenario is the most penalising, and I therefore consider this to be a minor shortfall.

939. The effect of uncertainties in the PAR performance are studied by:

- decreasing the rate coefficients in the recombination rate equation (described in sub-section 4.5.5.1) by up to 40%, and
- assuming that there is no PAR operating in the compartment where the break occurs.

940. The effect of reducing the recombination efficiency by 40% has a fairly small effect on the local peak hydrogen concentrations, because the PARs are not very responsive to the rapid increases in concentration that occur during the degradation phase. As expected, the hydrogen concentrations in the break compartments (the compartments into which the mass and energy releases occur, in this case the loop 2 RCP and PRZ compartments) and the containment dome decrease more slowly from their peak values when the efficiency is decreased, leading to higher concentrations. However, the long-term (>12 hours) concentrations are only very slightly increased. Counterintuitively, the safety case appears to show that the total amount of hydrogen removed by the PARs is slightly higher when the efficiencies are decreased, with a maximum at 80% efficiency (Ref. 198). The sensitivity study is also a fairly simplistic approach; no assessment is performed to consider if all of the hydrogen entering the PAR is consumed, or if any excess is allowed to exit. However, the results of the modelling indicate that even if these assumptions are shown to be not conservative, the resultant impact on the hydrogen concentration will not be significant. As such, I consider this to be a minor shortfall.

941. Deactivating the PARs in the break compartments has very little effect on the local hydrogen and global concentrations; the total amount of hydrogen recombined is decreased by about 14 kg (3% of the total generated). This suggests that, even when PARs are operating at full capacity, the decrease in hydrogen concentration in the vicinity of the break almost all occurs by physical transport out of the compartments to the containment dome.

4.5.5.3 Strengths

942. The RP has produced a coherent safety case, that takes appropriate account of all of the hydrogen production mechanisms.

943. The sensitivity studies on PAR operation show that long-term hydrogen concentrations in the containment would be suppressed even with substantially lower operating efficiencies.

944. The chemistry assumptions used to calculate the hydrogen challenge in DBC and DEC analysis are reasonable and conservative.

4.5.5.4 Outcomes

945. Based upon the assessment of the chemistry aspects of Hydrogen Generation and Control in UK HPR1000 described in sub-section 4.5.5 above, I have not identified any Assessment Findings which need to be addressed by the licensee. I have raised three minor shortfalls.

4.5.5.5 Conclusion

946. The RP has demonstrated that the generation of hydrogen during accident scenarios has been adequately modelled and that the risks of hydrogen generation have been adequately mitigated by the generic design of the UK HPR1000.

947. Based on the outcome of my assessment of hydrogen generation and control in UK HPR1000, I have concluded that that the claims, arguments and evidence as presented in this area appear reasonable and believe that the RP has made an adequate case to support GDA.

948. I have not identified any Assessment Findings as a result of this part of my assessment.

949. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used the guidance contained in SAPs FA.1, which required severe accident analysis to be performed to demonstrate that risks are reduced SFAIRP. I used NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors and chemistry assessment, and NS-TAST-GD-007, which concerns Severe Accident Analysis (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the generic safety case.

4.6 Chemistry Limits and Conditions Necessary in the Interests of Safety

950. Limits and conditions necessary in the interests of safety (operating rules) are an integral output of any safety case in defining the boundaries of safe operation. The prime purpose of setting limits and conditions necessary in the interests of safety is to prevent operation in unsafe conditions and to limit the consequences of fault/accident conditions should they arise. The safety case should define what these conditions are and how the plant is operated and controlled to maintain within them. From a chemistry perspective this should include limits for the operational chemistry. In line with the expectations of SAP ECH.1, they should be clearly derived from the safety case and linked to any chemistry claims, with substantiation from OPEX or theoretical analysis.

951. PCSR Chapter 21 sets out the main chemistry claims and arguments for UK HPR1000 (Ref. 5). These are discussed in more detail throughout my assessment in Section 4, however many are related to the definition and justification of the chemistry regime (and therefore limits and conditions necessary in the interests of safety) for the systems important to safety. The PCSR Chapter defines limits and conditions necessary in the interests of safety to include the following parameter types:

- Control: to control the plant and require a corrective action if an associated limit is breached, which can be further sub-categorised by:
    - Operating window (expected range): normal operating control parameters. Appropriate actions should be taken to recover the normal operating value within a specific time in the case of deviation.
    - Limits: inputs or initial conditions in fault studies. The outermost limits for safe operation and beyond which plant or research data or

engineering judgment indicates that it is inadvisable to continue to operate.

- Diagnostic: provides a means of monitoring plant performance, without limiting operations.

952. This type of approach is almost universally applied in PWR chemistry and is very similar to that proposed in both the IAEA and VGB guidelines; hence I consider this to be a suitable basis on which to develop limits and conditions necessary in the interests of safety (Ref. 9, Ref. 199).

953. The RP provided a document, ‘Generic Water Chemistry Specification (LCO)’ (Ref. 30), which summarises the water chemistry and radiochemistry limits and conditions necessary in the interests of safety proposed for the UK HPR1000, covering:

- the use and control of neutron poisons within the coolant;
- the integrity of SSCs important to safety, particularly those related to the fuel, pressure boundary or containment; and
- reduction of the generation and build-up of radioactive material to reduce radiation doses to personnel, to reduce the radioactivity of wastes and to reduce discharges to the environment.

954. Limits are proposed in the following operating modes:

- Reactor in power
- Normal shutdown with SGs
- Normal shutdown with RIS [SIS] in RHR mode
- Maintenance cold shutdown
- Refuelling cold shutdown
- Reactor completely discharged

955. The chemistry and radiochemistry limits proposed by the RP for power operations are replicated in Annex 3 of this report.

956. The RP notes that the ‘Generic Water Chemistry Specification (LCO)’ document is intended as a high-level summary of limits and conditions necessary in the interests of safety that are demonstrated in the underlying topic reports. I assess the justification for the specific limits proposed by the RP in the other parts of my assessment in Section 4, however, more generally I note that the list of limits and conditions necessary in the interests of safety provided in the LCO document is perhaps more detailed than expected at this stage of the development of the UK HPR1000. This is welcome and demonstrates the depth of understanding that the RP has developed in defining the chemistry regime. The RP has also proactively shared a significant amount of detail about the proposed limits and operating ranges in the underlying topic reports. Although these values may change during later phases of the safety case development, they provide a useful basis for a judgement on the overall chemistry regime, confidence that the regime has been adequately considered, and a basis for the licensee to use to adequately define all relevant limits and conditions necessary in the interests of safety. Further information on some of the preliminary limits and conditions necessary in the interests of safety is described in sub-section 4.2.9 of my report in terms of the holistic approach to primary circuit chemistry.

957. During construction and commissioning of a UK HPR1000, a detailed chemistry document, containing the basis and description of the chemistry requirements for the various SSCs through all phases of operation, for example, purity, cleanliness or material compatibility, will be necessary. Consideration of the required documentation has been presented in the ‘Topic Report on Commissioning Chemistry’ (Ref. 121). This proposed the development of a number of documents that take the form of both Test

Procedure and Test Guidance at each commissioning substage, which feed into the overarching System Commissioning Programme. The proposed chemistry requirements presented are largely centred around the cleanliness guidelines within RCC-M for its corrosion resistant or non-corrosion resistant material classifications in primary, secondary and auxiliary circuits (Ref. 121, Ref. 54). Chemical specifications (limits and conditions necessary in the interests of safety) are essential for all phases from component fabrication onwards and also include the preventative exclusion of foreign materials, reagent purity, guides to operating make-up and purification systems and expected surveillance and cleanliness during fabrication, commissioning and in normal operations. Such guides and specifications would make the chemistry in the various systems transparent to all involved and any problems should be avoided. Production of this type of documentation should be pursued at a later stage by the licensee, which I consider to be part of normal business in the site-specific phase. Further assessment of the commissioning chemistry proposed for UK HPR1000 is described in sub-section 4.2.11.

4.6.1 Strengths

958. The RP has described a suitable basis on which to develop the chemistry related limits and conditions necessary in the interests of safety for UK HPR1000, including the initial identification of many of the limits expected to be included in a future, more developed safety case.

4.6.2 Outcomes

959. Based upon the assessment of chemistry limits and conditions necessary in the interests of safety in UK HPR1000 described above, I have not identified any Assessment Findings or minor shortfalls.

4.6.3 Conclusion

960. Based on the outcome of my assessment of chemistry limits and conditions necessary in the interests of safety in UK HPR1000, I have concluded that that the basis for developing the chemistry-related limits and conditions necessary in the interests of safety as presented in this area appears reasonable and believe that the RP has made an adequate case to support GDA.

961. As part of my assessment of this aspect of the RP’s case I took account of the expectations set out in the SAPs, in particular SAPs ECH.1 and ECH.3, which point to the need to adopt a systematic approach to identify the chemistry limits and conditions necessary in the interests of safety that need to be applied in the interests of safety (Ref. 2). I also referred to the guidance on expectations for chemistry limits and conditions necessary in the interests of safety contained within NS-TAST-GD-088 (Ref. 6).

4.7 Demonstration that Relevant Risks Have Been Reduced SFAIRP

962. The demonstration that relevant risks within the generic design of the UK HPR1000 have been reduced SFAIRP is a key part of the generic safety case, and for all of the different parts of the chemistry case that I have sampled in this assessment, it has been a key focus in this report.

963. Throughout Step 3, and at the start of Step 4 of GDA, the RP struggled to demonstrate in the safety case documents that risks had been reduced SFAIRP. The topic reports generally included a list of facts, or measures that had been undertaken, and then a conclusion was made that this meant that risks had been reduced SFAIRP. This approach meant that there was no exploration of further reasonably practicable measures that could be taken to reduce risks. Although it was sometimes clear that a

position had been reached where risks were reduced SFAIRP, this had not been adequately described in the case and therefore credit could not be taken for it.

964. During Step 4 of the GDA project, the RP made improvements to this aspect of the case. In particular, the RP has developed their arguments to consider further improvements to reduce risks further, and a discussion of why these improvements may or may not be practicable to implement. These additions help to provide a more descriptive argument about why the position reached at the end of the GDA project means that risks have been reduced SFAIRP.

965. Although general improvements have been observed, this improvement is not seen consistently across all of the safety case deliverables. The documents produced as a response to ROs have generally included a more comprehensive argument that risks are reduced SFAIRP. Other examples include the response to FAC risk reduction, which was produced in response to RO-UKHPR1000-0034, and the justification that fuel deposits have been reduced SFAIRP, which was produced in response to RO-UKHPR1000-0015 (Ref. 141, Ref. 103).

966. Two design modifications were included within the generic UK HPR1000 design which have relevance to chemistry; the inclusion of zinc injection to the primary circuit, and the change of operating hydrogen concentration in the primary circuit, (discussed in sub-sections 4.2.6 and 4.2.7 respectively. I judged that both of these modifications aligned with RGP, and therefore contribute to the demonstration that risks are reduced SFAIRP for the generic UK HPR1000 design.

967. Although I have not encountered any significant deficiencies in the wider safety case, it is apparent that the justifications in topic reports which are not covered by ROs are not as strong as those that were written in response to an RO. Although adequate for GDA purposes, the licensee should seek to strengthen the arguments and justifications that risks are reduced SFAIRP in these areas, to ensure that a coherent case is developed; a number of Assessment Findings have been raised to this effect. This conclusion is also apparent within PCSR Chapter 21 and its associated ALARP Demonstration Report (Ref. 5, Ref. 17). The RP decided to produce a separate justification document, rather than incorporate SFAIRP arguments into the PCSR. This means that a lot of system descriptions are repeated, and the arguments are diluted as a result.

968. The RP has made significant efforts to review the design, and justify that the risks are reduced SFAIRP, and made changes where appropriate. I am therefore content that the intent of the guidance set out in NS-TAST-GD-005 ‘Guidance on the Demonstration of ALARP’, to provide fit-for-purpose demonstrations that risks have been reduced SFAIRP, has been met (Ref. 6).

4.7.1 Strengths

969. For most areas of the generic safety case, a clear and comprehensive demonstration has been provided, that identifies the measures taken to reduce risks SFAIRP.

4.7.2 Outcomes

970. Based upon the assessment of the overall demonstration that risks have been reduced SFAIRP in UK HPR1000 described in section 4.7 above, I note that there are a number of areas of the chemistry safety case in which, although adequate for GDA purposes, the licensee should seek to strengthen the arguments and justifications that risks are reduced SFAIRP. These are recorded in a number of Assessment Findings which are summarised in Annex 2.

4.7.3 Conclusion

971. As part of my assessment, I used the guidance provided in the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-005, which provides guidance on the demonstration that risks have been reduced SFAIRP, and NS-TAST-GD-088 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the safety case.

4.8 Consolidated Safety Case

972. Throughout the latter stages of GDA, the RP has updated documents within the generic safety case to reflect changes caused by:

- Development of the evidence base
- Response to ROs and RQs

973. When changes have been made to one part of the case, it is important that this change is reflected in the wider safety case. It is also important that where the RP has indicated that a change will be made, such as in response to an RQ, that this is also captured in the wider safety case. I therefore sampled several documents from the different tiers of the safety case, to make a judgement of the adequacy of the RP's approach to this topic. These were:

- 'Pre-Construction Safety Report Chapter 21 – Reactor Chemistry – Revision 2', a Tier 1 document (Ref. 5).
- 'Topic Report on Power Operation Chemistry – Revision D', a Tier 2 document (Ref. 28).
- 'Topic Report on Zinc Injection in the Primary Circuit of UK HPR1000', a Tier 3 document (Ref. 106).

974. From my sample, I found that generally the responses to RQs and the output from RO closure activities had been successfully included within the Tier 3 documents, and these had also been summarised accurately within the higher Tier documents. Throughout Steps 3 and 4 of GDA, the RP regularly produced new revisions of the lower tier documents (Tiers 2 and 3), which meant that new information was regularly incorporated into the case, which avoided the need to make significant changes as Step 4 concluded. I am content that the major changes made during GDA, such as the inclusion of zinc dosing, have been captured within the generic safety case. I also reviewed PCSR Chapter 21, as significant changes have been made to the case since the previous revision of this document was issued; I was satisfied that it had adequately incorporated the changes, and that the final safety case is consistent with my assessment.

975. The UK HPR1000 PCSR is an important document in the overall safety case hierarchy (Ref. 5). It is the top-level document which summarises relevant information in the documentation which sits beneath it. The preceding sections of this report have assessed the chemistry claims made by the RP. This following 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 (Ref. 2, Ref. 6). My assessment focusses on whether the PCSR provides an appropriate summary of the key aspects of the detailed case which sits beneath, and adequately justifies why the generic UK HPR1000 design is safe, and relevant risks reduced SFAIRP.

976. The RP has clearly devoted significant time and resources during Step 4 of GDA to develop the quality of the PCSR, and as a result, there are many strengths to it. The

document contains all of the topics that I would expect to see, and draws out the main Claims, Arguments and Evidence for each topic. The PCSR uses a coding system to identify each claim, and the associated arguments and evidence, and this is consistently applied in the wider generic safety case. The narrative description in the PCSR also links in with the CAE structure, which results in a case that is straightforward to navigate. Chemistry is a cross-cutting topic, and the RP has provided links to other PCSR chapters which contain information relevant to the document.

977. Other aspects of the document are not as strong, and I expect these to be improved upon by the licensee. The PCSR does not describe the categorisation of safety functions and classification of SSCs that are relevant to chemistry, nor link to a description of them in the wider safety case; details can be found in the SDMs for items of plant, but this is not easy to navigate within the case. The RP made the decision to separate out the justification that risks have been reduced SFAIRP from the PCSR into a separate document, but as I indicated in a previous section, I do not consider this is effective, and introduces unnecessary repetition into the case. Additionally, despite the RP making considerable efforts through Step 4 of GDA to improve the arguments that risks have been reduced SFAIRP, these are not well-reflected in the ALARP document.

978. 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, from a chemistry perspective, I am satisfied that the PCSR is adequate for GDA (Ref. 2, Ref. 6).

4.8.1 Strengths

979. From my sample, I have gained confidence that updates have been successfully consolidated within the wider generic safety case, and that top tier documents adequately summarise the information set out in the lower tier reports.

4.8.2 Outcomes

980. Based upon the assessment of safety case consolidation in UK HPR1000 described in sub-section 4.8 above, I have not identified any Assessment Findings.

4.8.3 Conclusions

981. Based on the outcome of my assessment of the consolidation of the generic safety case, I have concluded that that the approach taken by the RP appears reasonable and believe that the RP has made an adequate case to support GDA.

982. As part of my assessment, I used the guidance provided in SAP SC.4, which requires a safety case to be accurate, objective and demonstrably complete for its intended purpose, and the SAPs ECH.1 and ECH.3, which require safety cases to consider all chemistry effects and for a demonstration that chemistry is controlled adequately, respectively (Ref. 2). I also used NS-TAST-GD-005, which provides guidance on the demonstration that risks have been reduced SFAIRP, and NS-TAST-GD-88 and NS-TAST-GD-089, which concern the chemistry of operating civil nuclear reactors, and chemistry assessment (Ref. 6). I am content that overall, the expectations set out in this guidance have been met by this part of the safety case.

4.9 Comparison with Standards, Guidance and Relevant Good Practice

983. The standards, guidance and RGP used in my assessment are referenced in context throughout Section 4, and are listed in sub-section 2.4 and in Annex 1. A summary of my judgement against the most relevant of these is as follows:

- SAPs ECH.1 to 4 relate to the chemistry aspects of safety cases. I am satisfied that the RP has appropriately considered these expectations during GDA, in particular with regard to ECH.1 on systematically examining the chemistry effects important to safety. I consider that further justification is required for some of the chemistry control and monitoring aspects (ECH.3 and ECH.4) defined by the generic safety case, but I am content that the intent of these SAPs has been considered appropriately for GDA.
- SAPs SC.2 to 4 and SC.5 relate to the production of an adequate safety case. I am content that the RP has met the intent of these assessment principles to the extent expected for GDA. I have also gained confidence that appropriate consolidation of the chemistry safety case has been achieved, as noted in sub-section 4.8. There are a number of areas where further work will be required by the licensee. These are captured in a number of Assessment Findings, but the generic case presented represents a suitable basis for this future development and broadly meets my expectations for GDA.
- SAPs EAD.1, EAD.2, EMC.13 and EMC.16 relate to material ageing and degradation and the adequacy of metallic components and structures. As described throughout this report, there is a key relationship between materials integrity and operating chemistry control. I consider that this relationship is adequately reflected in the RP’s chemistry submissions, which provide a structured consideration of these matters. Although there are several related areas where additional development is required, as recorded in a number of Assessment Findings, these can be satisfactorily resolved by the licensee.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

984. This report presents the findings of my Chemistry assessment of the generic UK HPR1000 design as part of the GDA process.

985. Based on my assessment, undertaken on a sampling basis, I have concluded the following:

- In general, the scope, structure and content of the generic safety case meet my expectations for this stage of the project. From a Chemistry perspective, an adequate PCSR has been produced, which provides a summary of and links to the underlying evidence. However, further work will need to be undertaken by the licensee to develop the chemistry aspects of the safety case; this is captured in a number of Assessment Findings.
- The RP has identified a suitable set of claims on the operating chemistry for all modes of operation. In the majority of cases, the RP has provided sufficient supporting evidence to demonstrate that the claims can be achieved by the generic design.
- The RP has provided an appropriate demonstration that the generic plant design and engineering is adequate to achieve effective control of chemistry for relevant systems (including dosing, monitoring and clean-up) and to maintain the chemistry within the limits defined within the safety case.
- The different chemistry requirements likely to be necessary during different operating modes, and during different stages of the plant's lifetime have been suitably considered. The chemistry regime appropriately balances competing effects including corrosion risk, operator dose, and radioactive waste generation. The major chemistry parameters which would be expected to form part of the plant operating rules have been identified, and while further work will be required by the licensee to fully define and substantiate the associated limits, the basis of the approach is acceptable.
- The RP has appropriately considered the through-life performance of the chemistry related SSCs, and the effects of the chemistry regime on the susceptibility to material degradation mechanisms in making suitably justified materials selection decisions.
- The generation, transport and accumulation of radioactivity has been adequately analysed and quantified, and the safety case has been successfully developed to provide an adequate demonstration that radioactivity has been reduced SFAIRP.
- An adequate demonstration has been made that the effects of chemistry during fault conditions, including the generation, transport and behaviour of radionuclides and reactive species, are understood and that risks have been reduced SFAIRP.
- For those risks on which chemistry can have an influence, an appropriate overall demonstration has been made that chemistry effects are understood and that risks have been reduced SFAIRP.
- Collectively, the documentation contained within the MDSL provides an appropriate starting point for the licensee to develop the UK HPR1000 safety case into the site-specific stage.

986. As a result of my assessment, I raised 21 Assessment Findings. These require licensee input or decisions to resolve and are therefore matters for the licensee to consider and take forward in its site-specific safety submissions; they do not undermine the UK HPR1000 generic safety submissions or design.

987. As noted in the main body of this assessment report, I have assessed the RP's submissions against the expectations set out in the SAPs and the additional guidance

set out in a number of supporting TAGs (Ref. 2, Ref. 6). The SAPs of most relevance to my assessment are set out in Annex 1. In general, and as explained in more detail in sub-section 4.9, I am satisfied that the RP has appropriately considered these expectations and has met the intent of the identified assessment principles to the extent expected for GDA.

988. Overall, based on my sample assessment of the safety case for the generic UK HPR1000 design undertaken in accordance with ONR's procedures, I am satisfied that the case presented within the PCSR and supporting documentation is adequate. On this basis, I am content that a DAC should be granted for the generic UK HPR1000 design from a Chemistry perspective.

5.2 Recommendations

989. Based upon my assessment detailed in this report, I recommend that:

- **Recommendation 1**: From a Chemistry perspective, ONR should grant a DAC for the generic UK HPR1000 design.
- **Recommendation 2**: The 21 Assessment Findings identified in this report should be resolved by the licensee for a site-specific application of the generic UK HPR1000 design.

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Annex 1

Relevant Safety Assessment Principles Considered During the Assessment

SAP No	SAP Title	Description
Engineering principles: Chemistry		
ECH.1	Safety cases	The safety case should identify and analyse how chemistry can impact safety during normal operations and in fault and accident conditions and demonstrate how the chemistry will be controlled. A systematic approach should be adopted that identifies the limits and conditions (operating rules) that need to be applied in the interests of 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: 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.

SAP No	SAP Title	Description
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).
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.
Fault analysis		
FA.1	Design basis analysis, PSA and severe accident analysis	Fault analysis should be carried out comprising suitable and sufficient design basis analysis, PSA and severe accident analysis to demonstrate that risks 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.
AV.2	Calculation methods	Calculation methods used for the analyses should adequately represent the physical and chemical processes taking place. Where possible, the analytical models should be validated by comparison with actual experience, appropriate experiments or tests.

SAP No	SAP Title	Description
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.
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.
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: Control of nuclear matter		
ENM.3	Transfers and accumulation of nuclear matter	Unnecessary or unintended generation, transfer or accumulation of nuclear matter should be avoided.
Engineering principles: Heat transport systems		
EHT.5	Minimisation of radiological doses	The heat transport system should be designed to minimise radiological doses.
Engineering principles: Integrity of Metal Components and Structures		
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.
EMC.16	Contamination	The potential for contamination of materials during manufacture and installation should be controlled to ensure the integrity of components and structures is not compromised.

SAP No	SAP Title	Description
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

Assessment Findings

Number	Assessment Finding	Report Section
AF-UKHPR1000-0011	The licensee shall, as part of developing the operational chemistry safety case, ensure robust operational plant data is included to support safety case claims and arguments. A methodology should be implemented to identify, select and use relevant data to underpin the safety case. The data should be used to demonstrate, amongst other matters, that the following are appropriate and achievable: ■ plant operability; ■ control of chemistry parameters; ■ representative sampling of relevant chemical species; and ■ limits and conditions.	4.2.2, 4.2.3, 4.2.4, 4.2.7, 4.2.8, 4.3.4, 4.3.5, 4.4.1
AF-UKHPR1000-0053	The licensee shall, as part of detailed design, justify that the design and operation of the hydrogenation station reduces relevant risks so far as is reasonably practicable.	4.2.2.4
AF-UKHPR1000-0112	The licensee shall, as part of detailed design, justify that the cobalt inventory within UK HPR1000 has been optimised, taking account of the relevant risks. The justification should include the impacts of different operating conditions and material choices on radioactivity, worker doses and component reliability.	4.2.3.1
AF-UKHPR1000-0113	The licensee shall, as part of detailed design, justify which valves will include cobalt-based hard facings. This should focus on those which do not fall into the “prohibited” category according to procurement specifications. The justification should demonstrate that relevant risks have been reduced so far as is reasonably practicable.	4.2.3.1
AF-UKHPR1000-0114	The licensee shall, as part of detailed design, demonstrate that the manufacturing routes of key structures systems and components reduce radioactivity so far as is reasonably practicable.	4.2.3.1

Number	Assessment Finding	Report Section
AF-UKHPR1000-0115	The licensee shall, as part of developing the operational chemistry safety case, demonstrate how operating practices have been optimised to reduce the risks associated with the generation, transport and accumulation of radioactivity, so far as is reasonably practicable. This should include the activities necessary during start-up and shutdown of the plant.	4.2.3.3, 4.2.10
AF-UKHPR1000-0117	The licensee shall develop the plant-specific corrosion product estimations, including all relevant factors, to justify that risks have been reduced so far as is reasonably practicable. The analysis should assess how the coolant chemistry and the plant design and operation, affect the generation and transport of all relevant corrosion products.	4.2.3.3
AF-UKHPR1000-0118	The licensee shall, as part of developing the operational safety case, define limits and conditions to ensure that the risks associated with tritium (^3H) have been reduced so far as is reasonably practicable.	4.2.3.3
AF-UKHPR1000-0119	The licensee shall justify the material choices of high strength components which are susceptible to stress corrosion cracking within UK HPR1000.	4.2.4.1
AF-UKHPR1000-0120	The licensee shall demonstrate the impact of the primary circuit hydrogen concentration on fuel deposit formation and behaviour in UK HPR1000, and justify that the proposed primary circuit hydrogen operating range reduces relevant risks so far as is reasonably practicable.	4.2.5.1
AF-UKHPR1000-0159	The licensee shall, as part of detailed design, justify the capability of the nuclear sampling system to perform representative sampling for zinc in the primary coolant.	4.2.8.1
AF-UKHPR1000-0160	The licensee shall justify the consequences of introducing feedwater with elevated dissolved oxygen levels into the steam generators, from the emergency feedwater system.	4.5.2

Number	Assessment Finding	Report Section
AF-UKHPR1000-0161	The licensee shall, as part of detailed design, demonstrate that all reasonably practicable improvements to the design and materials selection have been implemented to reduce the risk of flow accelerated corrosion so far as is reasonably practicable. This should include implementing the commitments identified during Generic Design Assessment, and minimising the reliance placed on chemistry mitigations, where it is reasonably practicable to do so.	4.3.4.2
AF-UKHPR1000-0162	The licensee shall justify the capability of UK HPR1000 to provide timely detection of condenser leaks, and shall review the requirements for automatic actions in the case of leaks that could impact nuclear safety.	4.3.5.1
AF-UKHPR1000-0164	The licensee shall demonstrate that the chemistry assumptions used in the steam generator tube rupture analysis are conservative, and provide a sensitivity analysis to demonstrate the significance of the chemistry assumptions.	4.5.2.2
AF-UKHPR1000-0165	The licensee shall ensure that the primary circuit activity limit of 5 GBq t^{-1} assumed during Generic Design Assessment is underpinned within the safety case, and shall justify any associated plant or operator actions.	4.5.2.2
AF-UKHPR1000-0166	The licensee shall justify the uncertainties assumed in the analysis of the behaviour of fission products during all relevant Design Extension Condition accidents.	4.5.3.1
AF-UKHPR1000-0167	The licensee shall demonstrate the effect that changes in temperature in the in-containment refuelling water storage tank have on iodine chemistry during severe accidents, and justify the consequences.	4.5.3.1
AF-UKHPR1000-0168	The licensee shall demonstrate that the pH control of the in-containment refuelling water storage tank accounts for all sources of acidification.	4.5.3.1
AF-UKHPR1000-0169	The licensee shall demonstrate that the source term associated with iodine production during severe accidents has considered all relevant scenarios, to demonstrate that the risks associated with iodine production have been reduced so far as is reasonably practicable.	4.5.3.1

Number	Assessment Finding	Report Section
AF-UKHPR1000-0170	The licensee shall, as part of detailed design, specify and justify the design of the containment filtration and exhaust system. This should include, but not be limited to: • defining all chemical requirements; • defining appropriate limits and conditions; • demonstrating that the design is capable of mitigating releases of radiation during operation of the unit; • demonstrating that the risk due to hydrogen combustion during accident scenarios has been mitigated; • justifying that the chemical effects relevant to the operation of the system have been considered as part of defining an adequate source term; and • demonstrating that risks have been reduced so far as is reasonably practicable.	4.5.2

Annex 3

UK HPR1000 Chemistry and Radiochemistry Limits during Power Operation

Primary Circuit

System	Control Parameter	Unit	Operating Window	Limit
Reactor Coolant System	Total Boron	mg kg^{-1}	As specified by fuel limits	As specified by fuel limits
	^{10}B Enrichment	at%	35-39	-
	Lithium	mg kg^{-1}	As specified by fuel limits	<3.5
	Hydrogen	cm^{3} kg^{-1}	[redacted]-50	10-50
	Chloride	mg kg^{-1}	0.15	1.5
	Fluoride	mg kg^{-1}	0.15	1.5
	Sulphate	mg kg^{-1}	0.15	1.5
	Zinc	µg kg^{-1}	[redacted]	[redacted]
	Nickel	µg kg^{-1}	Site-specific stage	6
	silica	µg kg^{-1}	0.6	1
	^{131}Ieq	GBq t^{-1}	Site-specific stage	16
	Σ_{gas}	GBq t^{-1}	Site-specific stage	[redacted]
	Δ^{133}Xe	MBq t^{-1}	[redacted]	-
	^{133}Xe/^{135}Xe	-	[redacted]	-
	Δ^{134}I	MBq t^{-1}	[redacted]	[redacted]

Secondary Circuit

System	Control Parameter	Unit	Operating Window	Limit
High Pressure Feedwater Heater System / Feedwater Flow Control System	$pH_{25\ °C}$		[redacted]	[redacted]
	Oxygen	µg kg^{-1}	<[redacted]	-
Steam Generator Blowdown System	Cation conductivity (25 °C)	µS cm^{-1}	<[redacted](1)	-
			<[redacted](2)	[redacted](2)
	Sodium	µg kg^{-1}	<[redacted](1)	-
			<[redacted](2)	[redacted](2)
	Total gamma	MBq t^{-1}	Site-specific stage	[redacted]
Main Steam System	^{16}N	kg h^{-1}	Site-specific stage	[redacted]

Note that 1) Power ≤25%; 2) Power >25%

Auxiliary Systems

System / Component	Control Parameter	Unit	Operating Window	Limit
Coolant Storage and Treatment System / Coolant Storage Tank	Oxygen	mg kg^{-1}	0.1	1
	Chloride + Fluoride		0.1	-
Reactor Boron Water and Make-up System / Boric Acid Storage Tank	Total Boron	mg kg^{-1}	7000-7700	-
	^{10}B Enrichment	at%	35-39	-
	Chloride	mg kg^{-1}	0.3	-
	Fluoride	mg kg^{-1}	0.3	-
Emergency Boration System / Emergency Boric Acid Tank	Total Boron	mg kg^{-1}	7000-7700	-
	^{10}B Enrichment	at%	35-39	-
	Chloride	mg kg^{-1}	0.3	-
	Fluoride	mg kg^{-1}	0.3	-
Safety Injection System / Accumulator	Total Boron	mg kg^{-1}	1300-1400	-
	^{10}B Enrichment	at%	35-39	-
	Chloride	mg kg^{-1}	0.15	10
	Fluoride	mg kg^{-1}	0.15	10
In-Containment Refuelling Water Storage Tank	Total Boron	mg kg^{-1}	1300-1400	-
	^{10}B Enrichment	at%	35-39	-
	Chloride	mg kg^{-1}	0.15	-
	Fluoride	mg kg^{-1}	0.15	-
Spent Fuel Pool / Fuel Pool Cooling and Treatment System	Total Boron	mg kg^{-1}	1300-1400	-
	^{10}B Enrichment	at%	35-39	-
	Chloride	mg kg^{-1}	0.15	-
	Fluoride	mg kg^{-1}	0.15	-

Auxiliary Systems

System / Component	Control Parameter	Unit	Operating Window	Limit
Containment Heat Removal System / Reactor Pit Flooding Tank	Total Boron	mg kg^{-1}	1300-1400	-
	^{10}B Enrichment	at%	35-39	-
Coolant Storage and Treatment System / Downstream of the boric acid evaporation unit	Total Boron	mg kg^{-1}	7000-7700	-
Component Cooling Water System	pH		[redacted]	[redacted]
	Total gamma	MBq t^{-1}	Site-specific stage	0.6
Gaseous Waste Treatment System	Hydrogen	v/v %	-	4
	Oxygen	v/v %	-	2
Nuclear Island Liquid Waste Discharge System	Site-specific stage			
Conventional Island Liquid Waste Discharge System	Site-specific stage			
Gaseous Stack	Site-specific stage			