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BWRX-300 UK Generic Design Assessment (GDA) Chapter 23 – Reactor Chemistry

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EXECUTIVE SUMMARY

This chapter of the Preliminary Safety Report (PSR) describes how the reactor chemistry regime will be implemented for the BWRX-300. It is based on well-established design, water chemistry and water purification approaches, materials selection and operating practices that have been developed and evolved over many years of Boiling Water Reactor (BWR) operational experience, all of which is considered to be relevant good practice.

Sufficient information is provided to allow regulatory assessment of the design and proposed chemistry regime to Step 2 of Generic Design Assessment (GDA). It is important to note that nuclear safety risks cannot be demonstrated to have been reduced As Low As Reasonable Practicable (ALARP) within the scope of a 2-Step GDA. However, the design aspects, as set out in this chapter will effectively contribute to the development of a future ALARP position. General Electric-Hitachi (GEH) will further develop the safety case, in the form of a Pre-Construction Safety Report (PCSR), in later steps of GDA and/or during site licensing stages.

Claims and arguments relevant to GDA step 2 objectives and scope are summarised in Appendix A, along with an ALARP position. Appendix B provides a Forward Action Plan, which includes future work commitments and recommendations for future work where ‘gaps’ to GDA expectations have been identified.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
AP	Activation Product
ARM	Area Radiation Monitoring
BAT	Best Available Technique
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CA	Cycle Average
CAE	Claims, Arguments, Evidence
CCS	Containment Cooling System
CD	Condensate Demineralizers
CF	Condensate Filter
CFD	Condensate Filter and Demineraliser System
CFS	Condensate and Feedwater heating System
CHT	Cold Hydrostatic Testing
CMon	Containment Monitoring
CP	Corrosion Products
CR	Control Room
CRD	Control Rod Drive
CRN	Clinch River Nuclear
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DAC	Design Acceptance Confirmation
DB	Design Basis
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
DST	Deposit Source Term
DZO	Depleted Zinc Oxide
ECP	Electrochemical Corrosion Potential

Acronym	Explanation
EHC	Electro-Hydraulic Control
EME	Emergency Mitigation Equipment
EPRI	Electric Power Research Institute
EUST	End User Source Term
FAC	Flow-Assisted Corrosion
FAP	Forward Action Plan
FLEX	FLEXible Coping Strategy
FMCRD	Fine Motion Control Rod Drive
FP	Fission Products
FPC	Fuel Pool Cooling and Cleanup
FW	Feedwater
FWH	Feedwater Heater
GDA	Generic Design Assessment
GEH	GE-Hitachi Nuclear Energy
GEZIP	GE Zinc Injection Passivation
GNF	Global Nuclear Fuels
HCU	Hydraulic Control Unit
HEPA	High Efficiency Particulate Air
HFT	Hot Functional Testing
HP	High-Pressure
HPCP	High-Pressure Condensate Pump
HSI	Human System Interface
HVAC	Heating, Ventilation and Air Conditioning
HVS	Heating Ventilation and Cooling System
HWC	Hydrogen Water Chemistry
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	Isolation Condenser Cooling and Cleanup System
ICS	Isolation Condenser System
IGSCC	Intergranular Stress Corrosion Cracking
KKL	Kernkraftwerk Leibstadt
LfE	Learning from Experience
LP	Low-Pressure
LWM	Liquid Waste Management system
MCA	Main Condenser and Auxiliary

Acronym	Explanation
MCR	Main Control Room
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Lines
MSLB	Main Steam Line Break
MSLRM	Main Steam Line Radiation Monitor
MSR	Moisture Separator Reheater
MSRIV	Main Steam Reactor Isolation Valve
NBS	Nuclear Boiling System
NHS	Normal Heat Sink
NMCA	Noble Metal Chemical Addition
NWC	Normal Water Chemistry
NZO	Natural Zinc Oxide
OB	Outboard
OGS	Offgas System
OLNC	On-Line NobleChem™
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PCCPA	Passive Containment Cooling Pipe Array
PCCS	Passive Containment Cooling System
PCS	Primary Containment System
PCV	Pressure Containment Vessel
PCW	Plant Cooling Water
PER	Preliminary Environmental Report
PPS	Plant Pneumatic System
PREMS	Process, Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring
PrST	Process Source Term
PS	Process Sampling
PSR	Preliminary Safety Review
PST	Primary Source Term
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RFP	Reactor Feedwater Pump
RGP	Relevant Good Practice
RHX	Regenerative Heat Exchanger

Acronym	Explanation
RI	Regulatory Issue
RIV	Reactor Isolation Valves
RM	Realistic Model
RPV	Reactor Pressure Vessel
RW	Reactor Water
SA	Severe Accident
SAP	Safety Assessment Principle
SCC	Stress Corrosion Cracking
SCCV	Steel-plate Composite Cavity Vessel
SCDS	Safety Case Development Strategy
SDC	Shutdown Cooling System
SFAIRP	So Far As Is Reasonably Practicable
SJAE	Steam Jet Air Ejector
SMR	Small Modular Reactor
SoDA	Statement of Design Acceptance
SSCs	Structures, Systems, and Components
SSG	Specific Safety Guide
TAG	Technical Assessment Guide
TASS	Turbine Auxiliary Steam Subsystem
TB	Turbine Building
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TGSS	Turbine Gland Seal System
TSV	Turbine Stop Valve
USNRC	U.S. Nuclear Regulatory Commission
WGC	Water, Gas and Chemicals
WQS	Water Quality Specification

TABLE OF CONTENTS

EXECUTIVE SUMMARY .... **iii**

ACRONYMS AND ABBREVIATIONS .... **iv**

23. REACTOR CHEMISTRY **1**

- 23.1 Reactor Cooling Circuit System 3
- 23.2 Fuel Pool System 16
- 23.3 Plant Cooling Water System 18
- 23.4 Isolation Condenser Systems 20
- 23.5 Passive Containment Cooling System 23
- 23.6 Boron Injection System General System Description 24
- 23.7 Offgas System Catalytic Recombiner 25
- 23.8 Radioactive Source Term 27
- 23.9 References 68

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE **73**

APPENDIX B FORWARD ACTION PLANS **77**

APPENDIX C1 HISTORICAL EVOLUTION OF BWR CHEMISTRY **79**

- C1.1 Introduction 79
- C1.2 Normal Water Chemistry 79
- C1.3 Hydrogen Injection and Noble Metal Addition 80
- C1.4 Zinc Addition 81
- C1.5 Control of Other Chemical Impurities 82
- C1.6 Conclusion 82

APPENDIX C2 CHLORIDE INGRESS MANAGEMENT **92**

- C2.1 Chloride Ingress Mitigation, Response and Recovery 92
- C2.2 Chloride Ingress Protection System 93

APPENDIX C3 COMMISSIONING CHEMISTRY **94**

APPENDIX C4 ACCIDENT CHEMISTRY **95**

LIST OF TABLES

Table 23-1: Reactor Water Quality Design (>10% Power) and Maximum Values................ 36

Table 23-2: Condensate Filter and Demineraliser Aystem Inlet Water Quality Design (>10% Power) and Maximum Values.. 37

Table 23-3: Feedwater Quality Design (> 10% Power) and Maximum Values..................... 38

Table 23-4: Reactor Water Quality During Operations > 10%, and < 10% Power, but >93oC... 39

Table 23-5: Fuel Pool Cooling and Cleanup Water Quality Normal and Maximum Values 40

Table 23-6: Water Quality for Demineraliser Makeup Water for Tanks................................ 41

Table 23-7: Isolation Condenser Pools Cooling and Cleanup System Design Values 42

Table 23-8: Reference BWR Radionuclide Concentrations in Reactor Water and Steam, Taken from ANSI/ANS-18.1-2020... 43

Table 23-9: Plant Parameters for Source Term Adjustment 46

Table 23-10: Realistic Model Noble Gas Reactor Steam Source Term 47

Table 23-11: Realistic Model Iodine Reactor Water/Steam Source Terms 48

Table 23-12: Realistic Model Volatile Fission Products Reactor Water/Steam Source Terms 49

Table 23-13: Realistic Model Activated Corrosion Products Reactor Water/Steam Source Terms.. 50

Table 23-14: Realistic Model Non-Volatile Fission Products Reactor Water/Steam Source Terms.. 51

Table 23-15: Realistic Model N-16 Reactor Water/Steam Source Terms............................ 52

Table 23-16: Realistic Model H-3 Reactor Water/Steam Source Terms 53

Table 23-17: Source Term Design Basis Parameters... 54

Table 23-18: Design Basis Noble Gas Reactor Steam Source Term.................................. 55

Table 23-19: Design Basis Iodine Reactor Water/Steam Source Terms............................. 56

Table 23-20: Design Basis Volatile Fission Products Reactor Water/Steam Source Terms.. 57

Table 23-21: Design Basis Non-Volatile Fission Products Reactor Water/Steam Source Terms.. 58

Table 23-22: Design Basis N-16 Reactor Water/Steam Source Terms 59

Table 23-23: Design Basis H-3 Reactor Water/Steam Source Terms 60

Table 23-24: Geometric Mean of BWRX-300:UK ABWR by Radionuclide Class for Realistic Model Source Term.. 61

Table 23-25: Geometric Mean of BWRX-300:UK ABWR by Radionuclide Class for Design Basis Source Term 62

Table A-1: Reactor Chemistry Claims and Arguments 75

Table B-1: Reactor Chemistry Forward Actions.. 77

Table C1-1: NWC BWR Technical Specifications and Fuel Warranty Limits 84

Table C1-2: Summary of Key Events in the Evolution of Water Chemistry 85
Table C1-3: Summary of Typical BWR Water Conditions Under NWC and HWC 86
Table C1-4: Comparison Between Earlier NMCA and OLNC 87
Table C1-5: Isotopic Composition of Natural Zinc and Depleted Zinc.................................. 88

LIST OF FIGURES

Figure 23-1: Flow Diagram Showing Main Systems of the Reactor Cooling Circuit 63

Figure 23-2: BWRX-300 Reactor Pressure Vessel and Internals 64

Figure 23-3: Main Steam Piping 65

Figure 23-4: Fuel Pool System 66

Figure 23-5: Part of the Offgas System Showing the Three Stage Offgas Recombiner 67

Figure C1-1: Chemical Dosing Systems 89

Figure C1-2: Plants Implementing NZO to DZO 90

Figure C1-3: Depleted Zinc Oxide Injection System 91

REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance

23. REACTOR CHEMISTRY

The purpose of this Preliminary Safety Report (PSR) chapter is to provide a functional description of the reactor chemistry management and control approach to be implemented for the BWRX-300. The BWRX-300 is the tenth generation of the Boiling Water Reactor (BWR) that incorporates the lessons learned in design, construction, operations, and maintenance from over 100 previous BWRs that have been built, operated, and in some cases, decommissioned. The BWRX-300 adopts modern design standards and an up-to-date chemistry management and control philosophy based on world-wide Relevant Good Practice (RGP) and Operational Experience (OPEX). Furthermore, the chemistry regime proposed for the BWRX-300 is essentially the same as that of the UK Advanced Boiling Water Reactor (ABWR), previously evaluated and endorsed by the UK regulators following Generic Design Assessment (GDA):

- Office of Nuclear Regulation (ONR): Step 4 Assessment of Chemistry for the UK ABWR, Assessment Report (Reference 23-1)
- EA and NRW: Generic design assessment of Hitachi-GE Nuclear Energy Limited's UK ABWR (Reference 23-2)

By adopting a holistic approach for materials selection, commissioning/operating chemistry and reactor operating practices, an optimised chemistry regime to support future demonstration that risks are reduced ALARP and employ Best Available Techniques (BATs) principles is presented.

Although focused on the main reactor cooling circuit, chemistry regimes are also described and justified for each of the relevant fluid containing systems, where chemistry management and control are important for maintaining plant integrity and safety function. In addition, this chapter provides a description of the chemistry injection and dosing systems, the cleanup/purification systems and the monitoring systems that measures key chemistry and radiochemistry parameters.

The Chemistry chapter interfaces with the following PSR chapters:

- PSR Ch. 4: NEDC-34166, “BWRX-300 UK GDA Ch. 4: Reactor,” (Fuel and Core) (Reference 23-3)
- PSR Ch. 5: NEDC-34167P, “BWRX-300 UK GDA Ch. 5: Reactor Coolant System and Associated Systems,” (Reference 23-4)
- PSR Ch. 6: NEDC-34168P, “BWRX-300 UK GDA Ch. 6: Engineered Safety Features,” (Reference 23-5)
- PSR Ch. 9A: NEDC-34171P, “BWRX-300 UK GDA Ch. 9A: Auxiliary Systems,” (Reference 23-6)
- PSR Ch. 10: NEDC-34173P, “BWRX-300 UK GDA Ch 10: Steam and Power Conversion Systems,” (Reference 23-7)
- PSR Ch. 11: NEDC-34174P, “BWRX-300 UK GDA Ch. 11: Management of Radioactive Waste,” (Reference 23-8)
- PSR Ch. 12: NEDC-34175P, “BWRX-300 UK GDA Ch. 12: Radiation Protection,” (Reference 23-9)

The Chemistry chapter interfaces with the following Preliminary Environmental Report (PER) Chapters:

- PER Ch. 5: NEDC-34222P, “BWRX-300 Radioactive Waste Management, Arrangements,” (Reference 23-10)

- PER Ch. 7: NEDC-34224P, “BWRX-300 UK GDA Radioactive Discharges,” (Reference 23-11).

The following UK and international regulations and guidelines are of relevance when considering development of the BWRX-300 chemistry regime:

- International Atomic Energy Agency (IAEA), Specific Safety Guide (SSG)-13: Chemistry Programme for Water Cooled Nuclear Power Plants (Reference 23-12)
- ONR, GDA Guidance to Requesting Parties (Reference 23-13)
- ONR, New Nuclear Power Plants: Generic Design Assessment Technical Guidance (Reference 23-14)
- ONR, Safety Assessment Principles (SAPs) (Reference 23-15)
- ONR, Technical Assessment Guide: Chemistry of Operating Civil Nuclear Reactors (Reference 23-16)
- ONR, Technical Assessment Guide: Chemistry Assessment (Reference 23-17)
- EA, GDA Guidance for Requesting Parties (Reference 23-18)
- EA, Engineering: Generic Developed Principles (Reference 23-19)
- Electric Power Research Institute (EPRI), BWR Water Chemistry Guidelines (Reference 23-20)

Claims and arguments relevant to GDA step 2 objectives and scope are summarised in Appendix A, along with an ALARP position. Appendix B provides a Forward Action Plan, which includes future work commitments and recommendations for future work where ‘gaps’ to GDA expectations have been identified.

23.1 Reactor Cooling Circuit System

23.1.1 General System Description

The reactor cooling circuit is a complex group of individual systems and components that recirculates water and steam through the reactor core and turbines. A useful schematic summary of the power cycle and Feedwater (FW) systems is provided in BWRX-300 CRN Turbine Heat Balance in Support of TVA PSAR (Reference 23-21), see Figure 23-1.

Reactor Pressure Vessel

The BWRX-300 Reactor Pressure Vessel (RPV) consists of the RPV with nozzles, integral Reactor Isolation Valves (RIVs) and other appurtenances, a removable closure head, the reactor internals, and supports; see Figure 23-2 taken from the “BWRX-300 Nuclear Boiler System, System Design Description,” (Reference 23-22). The RPV instrumentation that monitors the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for the Fine Motion Control Rod Drives (FMCRDs).

The RPV is a vertical, cylindrical pressure vessel fabricated with forged rings welded together, with a removable torispherical closure head by use of a head flange, seals, and bolting, and a torispherical bottom head. The BWRX-300 RPV steam volume and Isolation Containment System (ICS) heat removal capacity are large enough, so pressure relief devices are not required during a steam line isolation event. The RPV also includes penetrations, nozzles, and reactor internals support.

The Reactor Coolant Pressure Boundary (RCPB) provides a barrier against the release of radioactivity generated within the reactor. The reactor pressure boundary design pressure is 10.342 MPa gauge (1500 psig). The specific reference point for design pressure is the inside bottom invert of the bottom head, but all portions of the RCPB is designed to at least 10.342 MPa gauge (1500 psig). Nozzles are designed to higher than 10.342 MPa gauge (1500 psig) where there is a pressure drop between the nozzle and vessel.

The design temperature of the RCPB and reactor internal components is 314.4 °C (598 °F). This conservatively corresponds to saturated steam temperature at the design pressure of 10.342 MPa gauge (1500 psig) and is used unless the component temperature is limited to a lower temperature, such as for components external to the RPV.

An increased internal flow path length, relative to forced circulation BWR, is provided by a “chimney” in the space that extends from the top of the core (top guide) to the entrance to the steam separator assembly. The reactor vessel high aspect ratio permits natural circulation driving forces to produce abundant core coolant flow. The top guide, the chimney (barrel), chimney head and steam separator assembly are supported by a shroud assembly that extends to the bottom of the core. The major reactor internal components include the following:

- Core components (control rods and nuclear instrumentation)
- Core support structures
- Chimney
- Chimney head and steam separator assembly
- Steam dryer assembly

Main Steam System

The Main Steam (MS) subsystem is comprised of two Main Steam Lines (MSLs) that are routed from the Outboard (OB) Main Steam Reactor Isolation Valve (MSRIV) discharge through the OB Main Steam Containment Isolation Valve (MSCIV) out to the Turbine Stop Valves (TSVs), the equalising header, and the MSL drains, see Figure 23-3: Main Steam

Piping(Reference 23-22). MS piping from the OB MSRIV discharge to the low point of the piping just inside the Turbine Building (TB) is designed to have a negative slope such that any steam condensation drains outwards to the MS drain lines.

Non-Seismic Category requirements including all the connecting branch lines up to the various steam loads including their respective isolation valves. The MS equalising header supplies High-Pressure (HP) steam via branch piping to the Turbine Bypass Valves (TBVs), the Moisture Separator Reheater (MSR), the No. 6 Feedwater Heater (FWH), and the Turbine Auxiliary Steam Subsystem (TASS) (Reference 23-22).

The MS drain lines provide the ability to drain the condensate from the MSLs to the condenser in a controlled manner during startup, lower power operation (approximately 40% nuclear boiler rated for the bypass drain valves), normal power operation, and shutdown (Reference 23-22).

High-Pressure Turbine

Two MS TSVs and Turbine Control Valves (TCVs) admit steam to two inlet nozzles of the HP turbine. Exhaust steam from the HP turbine is directed to the MSR. Additional extraction steam flows to the MSR for first stage of reheating and to the No. 5 FWH and No. 4 FWH. The Turbine Gland Seal Subsystem (TGSS) supplies sealing steam from the equalising steam header to the turbine shaft/casing to prevent the escape of radioactive steam and to prevent air in-leakage through sub atmospheric turbine glands.

Moisture Separator Reheater System

The MSR system is supplied steam from the HP turbine exhaust and both dries and reheats the steam before redirecting it to the Low-Pressure (LP) turbines as described in the "BWRX-300 Moisture Separator Reheater System: System Design Description," (Reference 23-23). Steam exhausted from the HP turbine enters the MSR and is dried using moisture separators. Separated condensate is routed to No. 3 FWH, while the dried steam progresses through two reheating stages employing internal HXs. The first stage of reheating uses HP turbine extraction steam for the initial temperature boost, with the cooled extraction steam being routed to No. 4 FWH. The second stage of reheating uses MS drawn from the equalising header to boost the exhaust steam temperature further, with the cooled extraction steam sent to the No. 5 FWH.

Low-Pressure Turbines

Reheated steam exiting the MSR flows via intermediate stops valves and intercept valves to two LP turbines. The expanded LP turbine extraction steam provides heating to the No. 1A/B, No. 2A/B and No. 3 FWH stages. The exhaust steam is directed to the twin condensers. The TGSS provides LP turbine shaft/casing gland sealing.

Main Condenser and Auxiliaries

During power operation, the main condenser is the heat sink for the steam cycle. The main condenser consists of two shells located beneath the LP turbines and drains to cross-connected hotwells as described in BWRX-300 Moisture Separator Reheater System: System Design Description (Reference 23-24). The Main Condenser and Auxiliary (MCA) receives exhaust steam from the LP turbines and TBVs. The main condenser is also the collection point for other steam cycle relief valve discharges, drains, and vents. The main condenser condenses and de-aerates the inputs and provides holdup for N-16 decay. The condensed water is routed from the hotwell to the condensate pumps to be reused in the power cycle. Main condenser hotwell level is controlled to maintain the net positive suction head to the condensate pumps. The Circulating Water System (CWS) provides cooling water through the tubes in the main condenser to provide a means of condensing the steam, 006N7757, "BWRX-300 Moisture Condenser and Auxiliaries System: System Design Description," (Reference 23-24).

The main condenser also serves as a collection and removal point for air in-leakage and non-condensable gases, such as hydrogen and oxygen, entrained in the TBV exhaust or turbine exhaust steam. The Steam Jet Air Ejectors (SJAEs) are used to remove the air from in-leakage and non-condensable gases from the main condenser. This helps to maintain the designed backpressure and improve plant efficiency. There are two SJAEs, with one normally in service and the other in a standby condition. Steam is supplied to the SJAEs from the gland steam system via a tap off the MS supply header to the gland seal system (Reference 23-24).

Condensate and Feedwater Heating System

As described in the “BWRX-300 Condensate and Feedwater Heating System (CFS): System Design Description,” (Reference 23-25), the CFS supplies FW to the RPV at the required pressure, temperature, and flowrate to maintain RPV level in the required band. In addition, the CFS supplies water or cooling water to various interface systems and responds to signals from the instrumentation and controls systems. The CFS system also provides the necessary indication, isolation functions, and design redundancy to support the required system safety functions (Reference 23-25). The condensate portion of the CFS is designed to pump condensate from the main condenser hotwell through three stages of low-pressure FW heating to the suction side of the reactor feed pumps in Mode 1, 2, 3, and 4 (Reference 23-25).

Condensate pumps (one duty and one standby) located below the normal water level of the condensers, takes suction from one of the hotwells and routes the condensate to the Condensate Filter (CF) and Demineraliser system (CFD), where insoluble and soluble impurities are removed. On exiting the CFD, condensate flows through various Heat Exchangers (HXs) to cool other process stream systems, i.e., Reactor Water Cleanup System (CUW), gland steam condenser, Offgas System (OGS) condenser and SJAE condenser. Fifty percent of condensate flow is pumped through two LP FWH stages, No. 1A FWH and No. 2A FWH, located in the neck region of the A condenser, and 50% condensate flow is pumped through No. 1B FWH and No. 2B FWH, located in the neck region of the B condenser. LP turbine extraction steam is used to raise the temperature of the condensate. One hundred percent of the condensate flow is pumped through a third LP heating stage, No. 3 FWH, using MSR condensate to provide a further temperature boost.

Variable speed FW pumps (one duty, one standby), route the condensate/ FW through three successive HP FWH stages raising temperature and pressure appropriate for injection into the RPV via four FW nozzles. No. 4 FWH, No. 5 FWH and No. 6 FWH raise the temperature of the FW, use extraction steam from the HP turbine/first MSR reheater stage; second MSR reheater stage; and equalising steam header respectively (Reference 23-25).

The FW heater drains system in the BWRX-300 is fully cascaded, as shown in Figure 23-1, whereby No. 6 FWH drains to No. 5 FWH, which itself drains to No. 4 FWH, etc., continuing down to No. 1A/B FWH, which drain directly into the A/B condenser hot wells. Any steam and non-condensable gases are vented to the condenser hot well.

Condensate Filters and Demineralisers System

As described in the “BWRX-300 Condensate Filters and Demineralisers System (CFD): Design Description,” (Reference 23-26), the CFD purifies the condensate and Reactor Water (RW), as required, to maintain reactor FW purity. The CFD uses filtration to remove suspended solids, including CPs, and ion exchange resin to remove dissolved solids from various condensate sources of impurities including CUW, Condensate Storage Tank (CST), condenser tube leakage and other impurities (Reference 23-26).

The CFD system includes three filter vessels with high efficiency backwash-type filter elements arranged in parallel, complete with remote operated valves, piping, instrumentation, and filter backwash equipment as is usual and customary for filtration design. Three demineraliser vessels are included and arranged in parallel, complete with remote operated valves, piping, controls, instrumentation, and resin replacement equipment (Reference 23-26).

To support the CUW system, the CFD is utilised to filter and remove ionic impurities from the CUW water that is being supplied to the CFS downstream of the condensate pumps. Prior to condensate water entering the inlet header to the CFs, redundant temperature elements exist to automatically open the Condensate Demineraliser bypass (CD) valve if the temperature exceeds 53.3°C (128°F) to prevent damage to the resin media (Reference 23-26) .

Demineraliser influent and effluent conductivities are monitored by the “BWRX-300 Process Radiation and Environmental Monitoring System,” (PREMS) (Reference 23-27) to determine condenser tube leakage and resin exhaustion.

Reactor Water Cleanup System

The BWRX-300 Reactor Water Cleanup System: System Design Description,” (Reference 23-28) provides blowdown-type cleanup flow for the RPV during reactor power operating mode. Actual cleanup by filtration and ion removal is performed by the Condensate Filter and Demineraliser System (CFD). Additionally, the CUW provides an overboarding flow path to the condenser hotwell (condensate pump suction) or liquid radwaste directly from the RPV lower region (Reference 23-28).

The CUW consists of one train fed by two inlet nozzles located in the RPV. The train’s inlet is independently connected to RPV penetrations located at about the mid-vessel height and take inlet from nozzles located near the RPV bottom head. This piping up to the RIVs are a part of the Nuclear Boiling System (NBS). The inlet piping connects to the reactor vessel and combines inside containment to form one discharge line. This line is provided with a containment isolation valve where it penetrates containment. This valve receives signal from CUW leak detection system and closes upon a detected leak. In the steam tunnel CUW has a sample line for water chemistry and the primary interface with Shutdown Cooling System (SDC). The CUW continues through a Regenerative Heat Exchange (RHX) and a pressure reduction station. The RHX and pressure reduction station are designed to condition the water to acceptable temperatures and pressures for processing to the CFD or overboarding. The RHX is designed to recover heat back to the vessel. Downstream of the RHX and pressure reduction station, there is a secondary interface with the SDC. Discharge piping is connected either to a condensate line for the normal CUW function or routed to the condenser hotwell or liquid radwaste for overboarding (Reference 23-28).

During normal operation the flow of RW from the RPV to the CFS is driven by the pressure differential between the two systems.

Shutdown Cooling System

The primary objective of the BWRX-300 Shutdown Cooling System: System Design Description,” (Reference 23-29) is to provide a decay heat removal pathway when shutting down the plant). The SDC system comprises two independent pump and HX trains. These trains together provide redundant decay heat removal capacity such that each train is designed to remove 100% of decay heat as soon as 4 hours after reactor shutdown. The major components of each train are a pump and a HX, along with valves, piping, instrumentation and controls, and power inputs. The two trains operating in parallel provide the system’s full rated shutdown cooling performance. Bypass lines and valves are included around the tube side of each HX to allow bypassing of the HX for SDC functions such as reducing RPV thermal stratification. The SDC system is capable of operating at HPs and temperatures up to the reactor operating pressure and temperature, however SDC operation at these elevated conditions is not expected during normal operation (Reference 23-29).

Each train’s suction is independently connected to a separate ICS condensate return line outside of containment and downstream of the ICS containment isolation valves. Each SDC train's return piping is independently connected to a separate CFS FW line outside of the containment isolation valves. This arrangement helps to prevent short circuiting of the flow inside of the RPV since the SDC suction is taken from the ICS return nozzle which originates in the chimney region over the core and the SDC return flow utilises the CFS return which terminates outside this region in the downcomer area (Reference 23-29).

Miscellaneous Systems

Following purification by the CFD, condensate may be stored in the CST, where it can be later reused in a wide range of water makeup and miscellaneous condensate supporting services managed by the BWRX-300 Liquid Waste Management (LWM) System (Reference 23-30).

Purified condensate is also provided to the Control Rod Drive (CRD) system for the continuous purge flow needed for the cooling of the FMCRD mechanisms. This purge flow also serves to prevent foreign material ingress from the bottom head of the RPV back into the FMCRDs. Control of the FMCRD purging functions and charging of the Hydraulic Control Unit (HCU) accumulators needed for hydraulic actuated reactor scram is provided by the CRD pump.

23.1.2 Chemistry Requirements

The BWRX-300 uses high purity water as the fluid to transfer thermal energy. The quality of water to which the reactor cooling circuit component is exposed, to a large extent, determines the reliability and lifetime of those components. Generally, corrosion effects stem from the combination of materials selected, the stresses or duty applied, and the environment to which the component is exposed.

Material corrosion caused by the environment is a function of the impurity and its concentration, temperature, and exposure time. Water quality is determined by materials selection, minimising the introduction of contaminants, generation of CPs, and the application of appropriate water treatment equipment to reduce contaminants introduced through operation and maintenance or from system materials.

Many decades of BWR operating experience have indicated that Intergranular Stress Corrosion Cracking (IGSCC) is the primary degradation mechanism that can affect the integrity of the stainless steel and nickel alloy internals and piping. Although water quality and design requirements cannot completely prevent IGSCC, maintaining the lowest practically achievable impurity levels through design minimises initiation and the rate of progression. Following water quality and design requirements aimed at minimising IGSCC also minimises the potential for fuel cladding corrosion. The chemical means for the control of IGSCC are Hydrogen Water Chemistry (HWC) and On-Line NobleChemTM (OLNC), see Section 23.1.3. Flow-Assisted Corrosion (FAC) of the steam side/condensate pipework is addressed by material selection and to a lesser extent by chemistry control of FW oxygen. Radiation buildup is also minimised by reducing corrosion and, therefore, deposition of corrosion particles on fuel that can become activated in the core.

As noted above, the reactor cooling circuit is a complex group of individual systems and components distributed across the Reactor Building (RB) and TB. Since each individual system/component forms a connected system group, a single integrated chemistry management and control approach is adopted for the entire reactor cooing circuit.

Based on evolution in BWR design, original EPRI water chemistry guidelines (Reference 23-20), past operational experience and current RGP, the reactor cooling circuit chemistry management and control philosophy for the BWRX-300 will be based on meeting three requirements:

Requirement 1: Minimise fuel pin leaks by maintaining cladding integrity

Requirement 2: Maintain structural integrity to all parts of the reactor coolant circuit and ancillary systems

Requirement 3: Reduce the radioactive source term to minimise:

a. dose rates to workers and the public

b. radioactive waste accumulations

c. releases of radioactivity to the environment

However, it is evident that accomplishing any one of the above requirements may give rise to an adverse effect to one of more of the others. For example, the injection of hydrogen into RW to suppress IGSCC is beneficial from the structural integrity perspective and thus complies with Requirement 2. Conversely, the same elevated hydrogen concentration also results in an increase in N-16 carryover into the MSLs, turbines and condenser systems leading to higher dose rates to workers and the public, contravening Requirement 3a.

The BWRX-300 chemistry management and control approach proposed for the reactor cooling circuit will, therefore, balance the potential conflicts between different requirements to achieve an optimised regime that reduces overall risks ALARP. Reactor cooling circuit chemistry will be managed and controlled by:

- Chemistry dosing
- Water purification
- Setting limits and conditions for chemistry parameters
- Sampling and measurement of chemical and radiochemical parameters
- Materials selection

These are discussed below.

23.1.3 Chemistry Dosing

BWRs were originally designed to operate under Normal Water Chemistry (NWC), with the chemistry controlled by the condensate and reactor CUW systems without chemical additives. In the decades of operating experience since then, advancements and operating experience have been made in understanding and improving water chemistry effects by controlling the chemistry via chemical additions. The planned BWRX-300 chemistry regime includes the following:

- **HWC** – Addition of hydrogen to the FW, resulting in excess hydrogen and therefore lower oxidants in the reactor coolant, for mitigation of IGSCC
- **OLNC** – Addition of platinum in conjunction with HWC during power operation to achieve low dissolved oxygen and surface potentials on BWR wetted surfaces
- **Zinc Injection, formally called GE Zinc Injection Passivation (GEZIP)** – The addition of Depleted Zinc Oxide (DZO) into the FW system to reduce Cobalt-60 deposition on out of core components

Hydrogen Water Chemistry

In-core radiation from fuel bundles causes radiolysis, which breaks down the water molecules into (mainly) hydrogen and oxidants (oxygen and peroxide). At full power, measured dissolved oxygen concentrations of RW are in the range of 150 - 300 ppb without hydrogen addition. This creates an oxidising environment which contributes to crack initiation and crack propagation growth rates in austenitic piping and reactor internals called IGSCC. The addition of excess hydrogen to the RW promotes the recombination of hydrogen and oxygen, as well as peroxide suppression. This was initially started in BWRs back in the 1980s. Excess hydrogen must be added to the RW for the recombination reaction to be effective and produce a more reducing environment as discussed in the "BWRX-300 Operating Water Chemistry Regime Design Basis Record," (Reference 23-31).

This can be quantified by measuring the Electrochemical Corrosion Potential (ECP) of the piping and internal surfaces. A reduction in ECP corresponds with reduced crack initiation and crack growth rates. Industry models use measured chemistry data to predict the chemistry and ECP in other areas of the reactor that are not measured. The target for ECP, in combination with excess hydrogen in the RW, is <-230 mV relative to the standard hydrogen electrode, at which corrosion mitigation is considered achieved.

The HWC system will maintain a constant FW hydrogen concentration by load-following, and the hydrogen addition rate will follow the reactor power level. Adjustments can be made if additional hydrogen concentration is needed during operating cycle based on monitoring various chemistry parameters. HWC is utilised prior to power operation when SJAE are in service, typically <5% power.

The hydrogen gas used for HWC will be continuously generated by an on-site electrolytic hydrogen generation system or gaseous hydrogen stored in HP tanks or trailers.

Under HWC water chemistry the composition at the inlet to the OGS becomes rich in hydrogen relative to oxygen. This can lead to explosion and/or flammability hazards, so is mitigated by bleeding air into the OGS to reestablish a stoichiometric hydrogen-oxygen mixture for catalytic recombination, see Section 23.7.

On-Line NobleChem^{TM}

OLNC is the injection of noble metals developed by GEH as a process to add catalyst to reduce the amount of bulk water hydrogen needed to achieve mitigation of IGSCC. While mitigation can be achieved with hydrogen injection alone, the bulk water hydrogen concentration required for mitigation is high enough to cause significant (~ 5X) increase in Main Steam Line Radiation Monitor (MSLRM) activity. Once the reactor internal surfaces have been coated with noble metals (platinum), the hydrogen/oxygen recombination reaction on internal surfaces is catalysed, requiring a much lower bulk hydrogen concentration to maintain a similar level of IGSCC mitigation. Lower hydrogen concentrations therefore result in essentially none to 10% increase in MSLRM activity. OLNC is performed during normal plant operation, specifically when power is >70% and core flow is >85% and with the HWC system in service. Platinum in the form of Na2Pt (OH)6 is added to the FW over a 10–14-day period and is applied approximately annually to deposit over the wetted surface area of the reactor vessel. Due to current fuel warranty limits, OLNC applications cannot occur until after 60 cumulative operating days after startup (with power > 10%) have elapsed following a refuelling outage. This timing allows any new fuel to be pre-conditioned prior to being exposed to the noble metal addition. The total amount of noble metal injected is determined by fuel surface area and the fuel warranty limits at the time of the application. Cumulative noble metal exposure is tracked over the lifetime of the fuel to ensure a maximum loading potential on fuel surfaces is not exceeded (Reference 23-31).

Applying at power allows for penetration into small crevices or difficult to reach areas by the noble metals due to higher temperatures and flow rates. This is important to prevent crack flanking. Crack flanking can occur during periods of time that HWC is offline. Without hydrogen injection, the oxidising nature of the water can cause cracks to grow at an unmitigated rate. If the crack grows beyond the influence of the applied noble metal zone the crack will continue to grow even when hydrogen is returned to service. A similar phenomenon can occur if the deposited noble metal is worn off in high flow areas or removed by outage activities. Applying noble metals more frequently is advantageous because any new cracks or areas where the noble metal wears thin (esp. in high flow regions) are soon recovered with a new coating of noble metals. Industry data have shown that OLNC + HWC is effective in keeping the ECP low and mitigating IGSCC.

Depleted Zinc Oxide

The largest contributor to radiation buildup on external recirculation system piping and other out of core surfaces is from Co-60 in RW during power operation. Co-60 is responsible for >90% of the personnel occupational radiation exposure incurred during refuelling outages. Historically, plants that have 5-15 ppb soluble zinc in the RW, from natural sources like brass condensers or by adding zinc to the FW, have the lowest radiation buildup rates. All US BWRs currently add DZO, in a passive system that dissolves DZO in water that is returned to the FW system. DZO is zinc that is depleted of the naturally predominant isotope Zn-64. Zn-64 is easily activated by neutron exposure to become Zn-65 which can become problematic for shutdown dose rates. Therefore, the Zn-64 isotope is removed (to <1% by weight) in DZO. This system is in operation with the startup of the plant and stays in operation until plant shutdown except for any maintenance activity that is required during operation (Reference 23-31).

Zinc addition reduces shutdown dose rates by competing with cobalt for adsorption sites on the fuel and other surfaces, as well as stabilising fuel oxide layers. A high concentration of soluble zinc present can overwhelm soluble cobalt, giving cobalt less chance to incorporate into fuel crud where it is likely to be activated by exposure to neutron radiation and become Co-60.

Injected zinc is incorporated into fuel deposits and is an important component of the tenacious spinel structure that stabilises fuel oxide layers. The presence of zinc in the fuel deposits as zinc ferrite promotes stability of the deposits and helps to retain Co-60 on the fuel, as opposed to its release to the water to later deposit on out of core surfaces.

23.1.4 Water Purification

Corrosion of all reactor coolant circuit wetted surfaces, together with impurity ingress from a variety of internal and external sources can result in an aggressive environment that may challenge the integrity of fuel and reactor components and increase the radioactive source term. At all times when the condensate pumps are available, control of impurity content is provided by the CFD. The CFD supplies the dual function for cleanup of both condensate and RW. Condensate is pumped from the condenser hotwell directly to the CFD for processing. RW is routed from the RPV, via the CUW, to the CFD for purification.

Outside of power operation (Plant Modes 1) and Hot Shutdown (Plant Mode 3) if the condensate pumps are unavailable and as a result there is insufficient differential pressure to operate the CFD (i.e., when power is <20%) other systems are available to perform cleanup of RW. This ensures that cleanup is always available. Cleanup redundancy in the event the condensate pump is out of service is provided as follows:

- Plant Mode 1 – Power Operation: sufficient differential pressure to directly drive water through the CFD.
- Plant Mode 2 – Reactor Startup: RW can be redirected to the SDC where it can be overboarded via pumps to either the MCA, where the water will then be cleaned by the CFD, or the LWM, where filters, ion exchange beds and reverse osmosis skids are available. On the way up to power operation, the SDC comes off and the CUW has sufficient pressure to run directly to condensate.
- Plant Mode 3 – Hot Shutdown: sufficient differential pressure to directly drive water through the CFD. There is also the option for redirection to the SDC where it can be overboarded via pumps to either the MCA or LWM depending on the temperature.

- Plant Mode 4 – Stable Shutdown: at the point where the reactor temperature has cooled such that the differential pressure is insufficient to drive flow through the system RW can be redirected to the SDC as described in Plant Mode 2 or to the LWM and onto the CST.
- Plant Mode 5 – Cold Shutdown: can use SDC as described to MCA if available or LWM to CST.
- Plant Mode 6 – Refuelling: either the SDC can be employed if the fuel pool gate is installed as described in Plant Mode 2 or the RW can be redirected to the Fuel Pool Cooling and Cleanup (FPC) if the gate is not installed, and the reactor cavity is flooded. If the cavity drained equipment gates are installed the SDC can be used to route water to the LWM.

Details of the CUW and SDC can be found in (Reference 23-28) and (Reference 23-29) respectively.

Impurities in Condensate

Since the BWRX-300 employs a fully cascaded FW heater drains system, all extracted steam is ultimately returned to the condenser hotwell, resulting in 100% of the MS line flow being condensed and routed via the condensate pump through the CFD. MS will contain various non-active and radioactive impurities transferred from RW. This will include non-condensable gases/volatile species and a small proportion of suspended particulate and dissolved solids in droplet carryover. While most of the non-condensable gases and some volatiles are extracted for treatment by the OGS, particulate and dissolved solids entrained in the MS transfer to the condensate collected in the condenser hotwell. Because of the large surface area and sections where two-phase flow is prevalent, the FW heaters and drains may also be a major source of soluble and particulate corrosion product loading to the condenser.

Impurities in Reactor Water

Unlike other BWRs, the BWRX-300 CUW does not provide the cleanup function itself but relies on the CFD for removal of water impurities. The main function of the CUW is to cool and depressurise RW from the RPV. The CUW reduces the RW temperature using a RHX and reduces pressure using a breakdown orifice. The cooled and depressurised RW is then routed downstream of a condensate pump for mixing with condensate, followed by filtration and demineralisation in the CFD. During Plant Operational Modes 1, 2, 3, and 4, CUW provides the equivalent of one percent of FW nominal flow (Reference 23-28). RW contains non-condensable gases/volatile species, stable and active CPs, Activation Products (APs) and Fission Products (FP).

Other Sources of Impurity

In addition to the impurities generated by reactor operation, contaminants may be introduced into RW and condensate via other routes. Most notably, these include:

- Chloride ions (Cl-) arising from condenser tube leaks, where the CWS uses sea water as the Normal Heat Sink (NHS)
- Sulfate ions (SO42-) typically generated from the degradation of organic ion exchange resin beads or introduced with cooling water via a condenser leak

For the open water structures including the fuel pool, equipment pool and reactor well, dust deposition is a potential source of silica (SiO2), chloride and organic material contamination. During refuelling outage, these bodies of water are connected to the reactor cavity thus allowing transfer of impurities to RW.

Makeup water is not a notable source of impurity.

Impurity Removal

The CFD purifies the condensate and the RW, as required, to maintain reactor FW purity. The CFD uses filtration to remove suspended solids, including CPs, and ion exchange resin to remove dissolved solids from various condensate sources of impurities including CUW, CST, condenser tube leakage and other impurities (Reference 23-26). The CFD is a full flow system that consists of high efficiency backwash-type filters followed by mixed, deep bed demineralisers with replaceable ion exchange resin. The CFD in combination with the FPC system and LWM system provide cleanup of RW and condensate during all plant modes.

23.1.5 Chemistry Parameter Ranges and Limits

Water chemistry parameters and limiting values are provided for the different reactor cooling circuit systems in the “BWRX-300 Water Quality Specification,” (WQS) (Reference 23-32). A more detailed specification will be available for future GDA submissions in the form of the “BWRX-300 Primary Water Chemistry Guidelines,” (Reference 23-33). The WQS provides the water quality limits designated as maximum values above which the plant may compromise the integrity of the plant Structures, Systems, and Components (SSCs). Typically, the plant operates well below the maximum values but based on historical operating data transients may occur. The plant design needs to account for these transients so that materials and system components can operate effectively while the plant responds to the transient. The response to a transient may be to shut down but other actions may be required with the plant in operation to best mitigate the effects of the transient.

Maximum Values are design limits where SSCs are capable of functioning with water chemistry values at the maximum values for a limited amount of time. Mitigating actions are required within six hours of reaching the maximum values based on industry guidelines unless there is an engineering evaluation demonstrating continued operation results in less degradation. For design margin, SSCs are capable to function for up to 24 hours in response to an excursion. Cumulatively, over the life of the plant, the SSCs have the capability to operate at the maximum levels for 0.5% of the plant operating life (Reference 23-32).

Based on RGP and OPEX, operational ranges and limits will be set for all important chemical parameters. The specifications for RW, the inlet to the CFD and FW are provided in, Table 23-1, Table 23-2 and Table 23-3 respectively.

23.1.6 Sampling and Measurement

The BWRX-300 has a plant-wide Process, Radiation and Environmental Monitoring System (PREMS) for most sampling, monitoring and measurement is described in (Reference 23-27) and in NEDC-34171P (Reference 23-6). This is made up of the following subsystems:

- Process Radiation Monitoring (PRM)
- Area Radiation Monitoring (ARM)
- Containment Monitoring (CMon)
- Process Sampling (PS)

PRM provides radiation monitoring of the MS, turbine and MCA systems and OGS. Chemistry sampling, monitoring and measurement of the reactor cooling circuit systems is provided by the PS. PS collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance.

For processes requiring continuous (or frequent) monitoring, sample tubing is routed from the process stream to one of three automated sample panels. Sample conditioning equipment is provided within each panel for pressure, temperature, and flow adjustment. Online monitoring equipment is used to the greatest extent practicable, and provided for measurements such as conductivity, dissolved oxygen, dissolved hydrogen, and total organic carbons. Grab sample taps are provided at the panels for infrequent/diagnostic sampling, online analyser checks, and instances in which online analysers are unavailable. Test connections are provided for equipment calibration or installation of temporary analysers.

Instrument data is displayed locally at the panel, with alarms to notify personnel when measurements exceed preset limits. Instrument data is provided to the Distributed Control and Information System (DCIS) and made available for use with third party laboratory trending/reporting software. Only parameters critical to plant operation are displayed in the Control Room (CR), with alarms to notify personnel when measurements exceed preset limits.

The PS provides online chemistry monitoring for the following reactor cooling circuit systems (Reference 23-27):

- CRD – Sample is taken downstream of the CRD pumps and routed to an automated sample panel
- SDC (when SDC is in service) – Samples are taken downstream of the supply flow elements and routed to an automated sample panel
- CUW – Samples are taken downstream of the supply flow elements and routed to an automated sample panel
- MCA – The PS provides online chemistry monitoring for each of the hotwells and the waterboxes within the MCA. Samples are taken downstream of each condenser hotwell and routed to an automated sample panel
- CFS – The PS provides online monitoring and local grab sampling equipment for CFS chemistry monitoring. Samples are taken from the following locations and routed to an automated sample panel:
  - Upstream of the CFs
  - Downstream of the final FWH

  Samples are also taken downstream of FWHs at heater drains locations and routed to a local grab sample station.
- CFD – The PS provides online chemistry monitoring of the CFD. Samples are taken downstream of the:
  - CFs (header)
  - CD (header)
  - Each demineraliser

Each sample (a-c above) is routed to an automated sample panel.

For failed fuel detection, the PRM and PS provides radiation monitoring and sampling of the OGS (Reference 23-27). This comprises:

- One radiation monitor is installed on the Charcoal Adsorber Vault Heating, Ventilation, and Air Conditioning (HVAC). This monitor provides indication of potential OGS leakage within the Charcoal Adsorber Vault.
- One noble gas monitoring skid is installed upstream of the Charcoal Adsorber Vault, between the Refrigeration Dryers and Offgas Reheater. The skid includes an online noble gas analyser and grab sampling taps. The skid provides baseline radiological data for monitoring OGS performance and is also the primary point of monitoring/sampling for fuel leakage. Elevated radioactivity levels at this location are typically indicative of fuel failures within the core. Periodic grab sampling is performed for isotopic analysis and trending.

- One noble gas monitoring skid is installed downstream of the Charcoal Adsorber Vault, between the Offgas High Efficiency Particulate Air (HEPA) filter and final discharge to the Continuous Exhaust Air Plenum. The skid includes a triple modular redundant set of online noble gas analysers (to support automated control logic) and integral grab sampling taps.

23.1.7 Materials Selection

The various reactor cooling circuits systems in BWRX-300 design will ensure corrosion resistant materials are used to minimise maintenance and maintain water quality. Appropriate material selection is based on a number of factors including temperature, design codes, flow velocity, oxygen levels, system size, and flow path. Materials are selected to minimise APs, withstand corrosion mechanisms, minimise production of CPs that can deposit on the fuel and minimise impurities. Systems that have a direct flow path to the reactor vessel have more stringent material requirements than systems with filtration or no flow path to the reactor vessel.

Where austenitic stainless steels are used in areas with water temperatures of 93°C (200°F) or higher exist for sustained periods, the carbon content of austenitic stainless steels are controlled. Operating experience has shown components exposed to steam or condensate require materials resistant to flow accelerated corrosion or erosion due to low oxygen content. Use of low-alloy steels also reduces the introduction of impurities such as iron particulates.

Cobalt as a contaminant in stainless steels and nickel base alloys in the reactor vessel and internal components shall be controlled. In general cobalt content shall be limited to 0.05 wt% maximum. For components immediately around the core such as shroud, top guide, and core plate, cobalt shall be further limited to 0.03 wt% or less. For in-core components such as control rods, in-core monitors including dry tubes and high-pressure FW tubes, the limit for cobalt content shall be 0.03 wt% or less.

23.1.8 Chemistry Management and Control During Reactor Startup and Shutdown

Chemistry management and control during startup (Mode 2) and shutdown (Modes 3, 4 and 5) adopts a different chemical dosing regime and operates under different water quality specification values compared with power operation mode.

Chemistry Dosing

Hydrogen Water Chemistry

During initial startup and startup from an outage, not all systems are online as soon as the RW is >200°F, the temperature at which materials become susceptible to IGSCC. HWC injection system is put in service as soon as the FW and offgas systems are placed into service (at ~5% power).

Hydrogen injection ceases during shutdown when the CFS and OGS are taken out of service (again at ~5% power).

Zinc Injection

During startup the GEZIP system begins operation (heatup) when the Reactor Feedwater Pump (RFP) is placed in service. The FW system must have at least one FW pump in service to provide flow through the GEZIP system.

During shutdown, the GEZIP system will be removed from service by isolating the skid from the CFS and allowing the system to cool down to ambient temperature naturally.

On-Line NobleChem^{TM}

OLNC is not utilised during startup and shutdown.

Water Quality

Water quality specification values are given for RW during operation above 10% power. For RW operation below 10% power (but above 93oC), corresponding to early stages of startup and cold shutdown, less strict specifications are provided. Specifications for CFD inlet and FW are given only above 10% power conditions. The values for RW are summarised in Table 23-4.

23.2 Fuel Pool System

23.2.1 General System Description

The fuel pool, equipment pool and reactor cavity pool are located above the reactor. During normal operation, Operational Modes 1-5, the equipment pool, and reactor cavity pool volumes are merged as a single pool volume, with the fuel pool gate installed, as shown schematically in Figure 23-4. During refuelling, Operational Mode 6, the fuel pool gate is open, and all three pools function as a single volume.

Fuel Pool

The fuel pool is used to store spent fuel after removal from the reactor, and new fuel after delivery to the site and before core loading. An area in the fuel pool is allocated for spent fuel cask loading operations. The fuel storage racks interface with the refuelling platform fuel mast grapple, allowing for unobstructed installation and removal of fuel assemblies, 007N2056, “BWRX-300 Fuel Storage Rack,” (Reference 23-34).

The deep pit in the fuel pool provides storage for spent fuel removed from the core during refuelling activities. New fuel is staged in fuel storage racks located in the fuel pool. An area in the deep pit is used for loading of a spent fuel cask. Shielding is provided by a minimum fuel pool depth of 10 feet (3m) above the top of stored fuel assemblies.

The fuel racks hold a single batch of new fuel and at least five years of spent fuel and have space remaining to accept a full core of off-loaded fuel. The racks are designed to allow natural convection through the rack and fuel for the removal of decay heat under normal and abnormal operating conditions.

Equipment Pool

The equipment pool is adjacent to the reactor cavity pool and lies opposite the fuel pool. It holds the steam dryer and separator during the refuelling outage, this pool is capable of being isolated by a removable gate.

Reactor Cavity Pool

The reactor cavity pool is situated centrally between the fuel pool and the equipment pool, and directly above the reactor pressure vessel. During startup, power operation and shutdown plant modes, the Steel-plate Composite Cavity Vessel (SCCV) head provides a seal with the main containment vessel, and the reactor cavity pool is filled with water. During outage, the reactor cavity pool is drained and the SCCV head removed. The RPV head is removed, and the reactor well/reactor cavity pool flooded up. Refuelling operation may then begin.

Fuel Pool Cooling and Cleanup System

The primary function of the “BWRX-300 Fuel Pool Cooling and Cleanup: System Design Description,” (Reference 23-35) is to provide continuous cooling of the water volume in the fuel pool to remove decay heat from spent fuel, and to provide replacement coolant inventory from a variety of sources, to ensure spent fuel is kept cool and submerged until relocated for permanent storage. In addition, the FPC includes demineralisation and particulate filtration to maintain coolant quality and to reduce general area dose. The FPC can be aligned to provide cooling and cleanup to the reactor cavity and equipment pools, as necessary (Reference 23-35).

The FPC processes water from the three pools to maintain pool temperature and water quality within prescribed limits for the plant. Fuel pool water (and reactor cavity water) overflows weirs into two surge tanks that feed the FPC. Additionally, the FPC is used to provide cooling if the Passive Containment Cooling System (PCCS) is exchanging heat with the equipment pool, see Section 23.5.

23.2.2 Chemistry Requirements

Chemistry management and control in the fuel pool system is implemented to meet the following requirements:

Requirement 1: Maintain a high-level of pool water clarity to aid visibility of fuel assemblies and RPV internal components during handling and storage

Requirement 2: Maintain fuel cladding integrity during extended underwater storage

Requirement 3: Maintain a low fuel pool radioactive source term

Fuel pool chemistry will be managed and controlled by:

- Continuous purification of fuel pool water (and, as required, equipment pool and reactor cavity pool water) through the FPC to remove insoluble and soluble species
- Setting limits and conditions for chemistry parameters
- Sampling and measurement of chemical and radiochemical parameters
- Materials selection

23.2.3 Water Purification

The FPC can use two 100% capacity pumps. The pumps discharge to individual FPC trains with 100% particulate filtration and a single deep bed demineralizer capable of processing 50% of either the A or B train. Each train has two 50% capacity shell and tube HXs cooled by the Plant Cooling Water (PCW) system. The deep bed demineraliser can be bypassed based on plant conditions and operational needs.

Impurity Removal

Fuel pool system impurities will include CPs from the fuel, fuel pool, equipment pool and reactor cavity pool. For defective fuel, there is potential for fission product release into the water also. Since the pools are open, airborne debris can provide another source of impurity ingress. Back-washable filters will remove particulate material and the deep bed demineralizer will remove soluble species. The ion exchange beads in the FPC tank can be sluiced out and replaced, as required.

23.2.4 Chemistry Parameter Ranges and Limits

The FPC is designed to have the capability to maintain water quality at the system design values listed in the WQS (Reference 23-32). Total system water quality is expected to meet or be better than the system design values. Based on RGP and OPEX, design and maximum limits are set for chemical parameters in Table 23-5.

23.2.5 Sampling and Measurement

The PS system provides online chemistry monitoring of the FPC (Reference 23-32). Samples are taken downstream of the demineraliser, and upstream and downstream of each filter. Each sample is routed to an automated sample panel.

23.2.6 Materials Selection

Materials selection is based on achieving high chemical compatibility between the materials and the pool waters, which will be maintained at high purity. The fuel pool, equipment pool and reactor cavity pools are constructed of a steel and concrete composite. The liners are stainless steel or equivalent corrosion resistant material. Even though liner corrosion is not anticipated, a leak detection system is provided to detect and monitor fuel pool liner leaks. The containment closure head is made from stainless steel cladding fixed to the outer surface to protect it from water in the reactor cavity pool.

23.3 Plant Cooling Water System

23.3.1 General System Description

The "BWRX-300 Plant Cooling Water System: System Design Description," (Reference 23-36) performs the heat removal function for different reactor and turbine related HXs in the BWRX-300. The PCW system is cooled by the CWS flowing through the PCW main HX to work as a barrier against radioactive contamination of the circulating water in the event of a reactor related HX leakage. The PCW system also maintains a constant operating temperature for the PCW supply lines despite variance in the ambient temperature (Reference 23-36).

Plant Cooling Water System

The PCW system consists of 2 main piping distributions that provide cooling water to different interfacing system HXs. The first piping distribution consists of two 100% trains to have full redundancy in case of loss of one train. This piping distribution is linked to the cooling of the following interfacing systems:

- FPC HX
- ICC HX
- SDC HX
- Plant Pneumatic System (PPS) compressor aftercooler

The second piping distributions is mostly associated with the utilities for condensate pumping or power generation has one 100% train which provides cooling water to the following system interfaces:

- FW pump motors
- FW pump adjustable speed drives
- condenser pump motor
- vacuum pump skid
- generator cooler
- isophase cooler
- lube oil cooler
- Electro-Hydraulic Control (EHC) cooler

23.3.2 Chemistry Requirements

The chemistry control requirement of the PCW system is:

Requirement 1: Minimisation of corrosion of pipework to ensure system integrity, safety function and achieve design life targets

Chemistry will be controlled by:

- Using a N_2 purge system to control oxygen concentration in water
- In line instruments to measure dissolved oxygen and conductivity
- A radiation monitor to detect leakage into the PCW
- Materials selection

In addition, the chemical dosing pots will be designed for the case where a plant needs to add corrosion inhibitor or biocide.

23.3.3 Chemistry Dosing

Both PCW trains share a chemical feeding pot, which enables operators to inject corrosion inhibitors, pH control chemicals, biocides, or other chemicals, when needed.

23.3.4 Chemistry/Impurity Control

The PCW surge tanks have a piping interface with Water, Gas and Chemicals (WGC) system which provides the needed demineralised water to make up for any water loss in the system. The surge tanks include level transmitters to alert operators in case of level drop, which would indicate a leak in the system.

23.3.5 Chemistry Parameter Ranges and Limits

Limits are not yet set for dosing chemicals. Demineralised makeup water will meet the quality requirements presented in the WQS (Reference 23-32), as shown in Table 23-6.

23.3.6 Sampling and Measurement

The PS provides local grab sampling station PCW chemistry monitoring. Sampling points are included on each train (upstream of the PCW pumps and downstream of each FPC HX) to measure water conductivity, radioactivity, and water quality.

23.3.7 Materials Selection

The PCW will use a combination of stainless steel (e.g., surge tanks) and carbon steel (piping).

23.4 Isolation Condenser Systems

23.4.1 General System Description

When in operation, the "BWRX-300 Isolation Condenser System: System Design Description," (Reference 23-37) removes heat from the reactor coolant and rejects it to the environment, thereby condensing reactor coolant supplied as steam from the RPV on the tube side of the Isolation Condenser (IC) and then returning the condensate back to the RPV in a closed loop. Heat removed from the steam is transferred by the IC to the water in the pool, which is vented to atmosphere. The IC is placed at an elevation above the "BWRX-300 Primary Containment System: System Design Description," (PCS) (Reference 23-38) and above the steam source so that this process is driven passively by gravitational force.

The ICS consists of three independent trains, each containing a HX or IC that is submerged in a dedicated pool of water and is connected to the RPV by steam supply and condensate return piping. The complex of pools represents the ultimate heat sink for protecting the reactor core under any off-normal event in which the main condenser is not available, and the RPV becomes isolated. The pools and atmospheric vents are in the scope of the Isolation Condenser Cooling and Cleanup System (ICC).

The steam supply line between the RPV and the IC is normally open, and the condensate return line is normally closed, allowing the IC and condensate return piping to fill with condensate that is maintained at a subcooled temperature by the pool water and the containment air temperature during normal reactor operation. This is considered the standby mode or condition for the ICS.

The ICS is placed in operation by opening the condensate return line to the RPV. The subcooled condensate that is stored in the system during the standby state enters the RPV chimney interior providing additional subcooled inventory while also quenching steam and lowering pressure at the exit of the reactor core. Simultaneously, steam from the RPV enters the IC where it is condensed in the tubes and returned to the RPV in a continuous cycle. If the RPV conditions fall below the saturation point, the ICS enters an idle state until decay heat drives conditions back to saturation, automatically and passively placing the ICS back into operation (Reference 23-37).

Isolation Condenser System Cooling and Cleanup System

The BWRX-300 Isolation Condenser Pools Cooling and Cleanup System: System Design Description," (Reference 23-39) processes water from the three IC cubicle pools and surrounding outer pools to maintain water temperature within the prescribed safety analyses and administrative limits established for the plant. The ICC also purifies the IC Pool Water to maintain established plant water quality standards. Consequently, the ICC has overall responsibility for the water contained in the IC pool compartment structure, including responsibility for designed safety features such as pool atmospheric vents, long-term diverse and FLEXible coping strategy/Emergency Mitigation Equipment (FLEX/EME) makeup water replenishment, and unidirectional pool makeup conduits that supply makeup water from the outer pool segments to the inner pools during off-normal conditions.

The ICC includes two identical, 50% capacity trains, each equipped with a centrifugal pump and frame-and-plate HX to transfer energy from the IC pools to the PCW. The two independent ICC trains allow maintenance to be performed on one train during reactor operation without having to disable the entire ICC.

23.4.2 Isolation Condenser Cubicle Pools

The IC pools consist of three inner pools or cubicles in which the IC HXs reside and the surrounding three outer pool segments. Each pool is dedicated to one ICS train, contains one IC, and is located in the RB directly above the Pressure Containment Vessel (PCV). The IC inner pools are physically separated from one another by structural walls. The complex of IC pools provides the ultimate heat sink for protecting the reactor core for any off-normal event in which the main condenser is not available, and the RPV becomes isolated.

23.4.3 Chemistry and Water Quality Management

The ICC includes a single integrated, skid-mounted demineraliser to remove soluble impurities from the IC Pool Water. The demineraliser can be isolated to permit resin exchange and other maintenance activities to be performed during reactor and ICC operation. The demineraliser is isolated to prevent degrading the resin media if the HX discharge temperature exceeds the maximum recommended service temperature for the resin media. The demineraliser is also isolated during system preventive maintenance in which chemicals are injected into the process water to inhibit corrosion and biological activity in the IC pools and ICC components.

Non-condensable gas that may accumulate within an IC is either removed by purging during system standby or neutralized by catalytic recombination during system operation. In the standby mode non-condensable gas is continuously purged to the NBS MSLs from a high point in the steam supply line of an ICS train.

23.4.4 Chemistry Requirements

The chemistry management and control requirement of the ICS is:

Requirement 1: Minimisation of corrosion of pipework to ensure system integrity, safety function and achieve design life targets

Chemistry will be managed and controlled by:

- Dosing of specific chemicals
- Purification of ICS pool water through the ICC to remove insoluble and soluble species
- Providing makeup water to the system to maintain level
- Sampling and measurement of chemical parameters
- Materials selection

23.4.5 Chemistry Dosing

The ICC is equipped with a dosing pot for injecting chemicals for corrosion and biological control into the process fluid.

23.4.6 Impurity Control

ICS pool system impurities will include CPs from the ICS pool liner, ICS HX/pipework and any airborne debris. In the event of an ICS tube/HX leak, radioactive activation, corrosion, and FPs will enter the pool water. A removable cover plate with coarse-mesh foreign material exclusion screen on the Suction Surge Tank shall prevent large pieces or debris from entering the Suction Surge Tank and potentially blocking or partially blocking the main suction to the ICC.

23.4.7 Chemistry Parameter Ranges and Limits

The ICC is designed to maintain the system water quality at the design values listed in the WQS (Reference 23-32). The water quality from the cleanup system output is expected to be much better than the values listed in order to maintain the overall system water quality at or below the system design values. Based on RGP and OPEX, limits will be set for chemical parameters as shown in Table 23-7.

23.4.8 Sampling and Measurement

PREMS provides PS of water entering and exiting the Demineraliser. Three samples will be routed to the PS panel in PREMS. Conductivity will be monitored on the common inlet and outlet of the IX and the outlet of the filter storage pool.

23.4.9 Materials Selection

All ICC components, piping, valves, and instruments with surfaces in contact with IC Pool Water is made of Type 304/304L or 316/316L stainless steel or metals with equivalent corrosion resistance.

23.5 Passive Containment Cooling System

23.5.1 General System Description

The PCCS transfers heat from the containment to the reactor cavity and equipment pools to maintain containment pressure and temperature within the design limits during accident conditions. Heat transfer occurs from the containment to the PCCS by natural circulation and condensation. The PCCS functions for containment depressurization and heat removal are passive and do not require on-site or offsite electric power system operation (Reference 23-38).

The PCCS consists of three independent trains, each with a component called the Passive Containment Cooling Pipe Array (PCCPA), which contains a minimum of six vertical pipes connected with a top and bottom header within containment. The piping connected to headers is routed through containment, as well as through the equipment pool. The equipment pool communicates with the reactor cavity pool when the PCCS is required to be operable. In Hot Shutdown, Startup, and power operation, the reactor cavity equipment pool gate is removed, connecting the equipment pool and reactor cavity pool to provide a heat sink for PCCS. The PCCPA units are placed at locations that are not subject to jet loads or protected against them.

Heat transfer occurs from the containment to the PCCS by natural convection and condensation through the PCCS and containment dome to the subcooled water in the equipment pool. There are no active components or actuation signals required, as the PCCS is always in service.

Fuel Pool Cooling and Cleanup System

The Fuel Pool Cooling and Cleanup System provides water to the equipment pool and heat sink for PCCS.

23.5.2 Chemistry Requirements

Since the PCCS interfaces with the fuel pool, equipment pool and reactor cavity pool, chemistry requirements are determined by these systems, see Section 23.2.2.

23.5.3 Water Purification

The FPC will maintain the water purity in the PCCS, see Section 23.2.3.

Impurity Removal

FPC filters and demineralizers will remove insoluble and soluble impurities from the PCCS water, see Impurity Removal in Section 23.2.3.

23.5.4 Chemistry Parameter Ranges and Limits

The WQS defined for the fuel pool are applicable to the PCCS, see Section 23.2.4 and Table 23-5.

23.5.5 Sampling and Measurement

The process monitoring system for the fuel pool, equipment pool and reactor cavity pool will be used for sampling and monitoring of the PCCS water, see Section 23.2.5.

23.5.6 Materials Selection

300 Series Stainless Steel shall be used for all PCCS piping.

23.6 Boron Injection System General System Description

23.6.1 Boron Injection System

The “BWRX-300 Boron Injection System (BIS),” (Reference 23-40) is a manually initiated, independent means of shutting down the reactor in the event that the hydraulic and screw driven control rod insertion systems fail. BIS consist of a storage tank, test tank, injection pump, piping, valves and instrumentation and controls required to prepare and inject a neutron absorbing solution into the reactor and to test the system. The BIS will be manually initiated from the Main Control Room (MCR) as directed by the emergency operating procedures (Reference 23-40).

23.6.2 Chemistry Requirements

The chemistry management and control requirement of the BIS is:

Requirement 1: Injection of sufficient negative reactivity into the reactor coolant system to ensure shutdown from full power to cold 20°C subcritical state with no control rod motion

BIS chemistry will be managed and controlled by:

- Pre-mixing granular sodium pentaborate decahydrate ($Na_2B_{10}O_{16}10H_2O$) enriched in B^{10} (to 94%) with demineralised water and stored in a suitable sized tank

This is a standby safety system that will be periodically tested.

23.6.3 Chemistry Parameter Ranges and Limits

A quantity of neutron absorber will be injected into the reactor vessel that produces a minimum concentration equivalent to 660 ppm by weight of natural boron in the reactor core at 20°C. This concentration, along with an additional 25% (165ppm) added to account for leakage and imperfect mixing, will result in a total requirement of greater than or equal to 825 ppm. Injection of sodium pentaborate solution will take place into the condensate return line of the ICS “C” loop downstream of the two ICS condensate return valves which will provide a direct flow path into the reactor.

23.6.4 Sampling and Measurement

Instrumentation in the MCR will provide information to determine BIS storage tank solution level, solution temperature, system flow rate, system status, injection pump status and the solution level in the test tank during testing mode. In the MCR the alarm management system shall sound high or low alarms for storage tank solution temperature and level, and or alert the operator when the system is inoperable. Initiating the BIS from the Human System Interface (HSI) display opens the storage tank outlet valve and injection valve and starts the injection pump. Sodium pentaborate solution will be pumped until automatic pump shutoff occurs at tank zero level or manual shutoff is initiated.

23.6.5 Materials Selection

The materials and equipment used in the BIS components shall be based on a 60-year design life, including appropriate provisions for maintenance and replacement. Stainless steel (304 or 316) is used for the storage and testing tanks due to its corrosion resistance.

23.7 Offgas System Catalytic Recombiner

23.7.1 General System Description

The HWC system injects hydrogen into the suction side of the FW pumps to mitigate IGSCC of reactor internals. The excess hydrogen being injected drives the recombination of hydrogen and oxygen in the vessel core region, resulting in a reduction of oxidants in the reactor coolant.

Hydrogen addition to the FW results in an excess ratio of hydrogen to oxygen at the entrance to the OGS. In order to consume the excess hydrogen at this location, the HWC system also controls the injection of air upstream of the offgas recombiner to maintain a controlled percentage of oxygen at the exit downstream of the offgas recombiner. This requires a monitoring system in the plant OGS at the offgas recombiner exit, which will consist of hydrogen and oxygen monitors.

“BWRX-300 Offgas System: System Design Description,” (Reference 23-41) recombiner is a three section vessel consisting of a preheater section, catalytic recombiner section, and condenser section, as shown in Figure 23-5. The purpose of the offgas recombiner is to perform the functions of oxygen and hydrogen recombination and to condense the resulting steam for return to the main condenser (Reference 23-41).

The purpose of the preheater section is to raise the temperature of the offgas so that the water content reaches a superheated state, ensuring that it fully boiled and gaseous prior to entering the recombiner section. The catalytic recombiner section uses a precious metal catalyst to promote the recombination of radiolytic oxygen and hydrogen gases into water. The offgas condenser section provides the first stage of condensation of the dilution steam in the effluent mix and the additional water vapor resulting from the recombination of the radiolytic gases.

23.7.2 Chemistry Requirements

The chemistry management and control requirement of the OGS Catalytic Recombiner is:

Requirement 1: Recombination of hydrogen and oxygen into water to maintain plant water inventory and reduce hydrogen detonation risk

Offgas recombiner chemistry will be managed and controlled by:

- Air injection
- Catalytic recombination of hydrogen and oxygen

Air Injection

The HWC system shall modulate air injection upstream of offgas recombiner with control signal input from the OGS gas analysers to achieve maximum hydrogen recombination.

23.7.3 Chemistry Parameter Ranges and Limits

The HWC system shall control hydrogen injection to the CFS such that a maximum H2 concentration does not exceed 1% by volume of the total offgas effluent flow.

23.7.4 Sampling and Measurement

Hydrogen concentration, oxygen concentration, and moisture content is monitored at the outlet of the dryers. All three detectors are triplicated to allow to allow continued detection in the event of any single failure. The annunciators of high hydrogen concentration shall be provided locally, signalling a potential detonation risk, and in the MCR for corrective action to be taken. High hydrogen can trigger a closure of the remote isolation valves connecting to the Heating Ventilation, and Cooling System (HVS) discharge and open the recirculation path back to the MCA. Low oxygen concentration alarm shall be provided at the MCR and serve as an input to the HWC system to adjust the injection of hydrogen or air (containing oxygen) as needed.

Carbon monoxide monitoring is provided at the outlet of the Guard Bed and last charcoal bed to detect ignition of the charcoal within the beds. Radioactivity level in the OGS is monitored by PREMS at the outlet of the refrigeration dryers and just prior to release. In addition, there are high radioactivity level alarms in the MCR.

23.7.5 Materials Selection

The pipe material used shall be SA 106 Grade B Carbon Steel.

23.8 Radioactive Source Term

23.8.1 Introduction

Primary Source Term

Two source terms are presented in this section. The first source term is a Realistic Model (RM) used to predict expected long-term average concentrations of radionuclides in the primary and secondary fluid streams. The second is a conservative Design Basis (DB) model that utilizes an established GEH fuel clad defect model.

- The RM, which is based on available measured nuclide concentrations during normal operation at operating BWRs, is determined from the American National Standards Institute (ANSI)/American Nuclear Society (ANS) 18.1 (Reference 23-42) and is the source term model used to estimate liquid and gaseous effluent releases. The RM is also referred to as the normal operation source term.
- The second source term conservatively employs a GEH fuel clad defect model and serves as a basis for system and shielding requirements and is referred to as the DB coolant source term.

Other Source Terms

Although not discussed further in this section a number of other source terms have been developed to support different areas of the safety case, so are covered in more detail in relevant PSR chapters. In summary these include some additional Process Source Term (PrST) analysis to support shielding design and End User Source Terms (EUST) specifically derived to support radioactive waste assessments and effluent discharge analyses.

Other Source Terms explicitly out of scope for GDA Step 2 are:

- Cycle Average (CA) Source Term
- Deposit Source Term (DST)
- Transient source term
- All additional PrSTs and EUSTs (other than those discussed above)

GEH is, however, committed to undertaking further source term analysis as the design evolves and the BWRX-300 moves through additional regulatory assessments. Development of these additional source terms is presented in NEDC-34274P, “BWRX-300 UK GDA Forward Action Plan,” (FAP), in Appendix B, see also (Reference 23-43).

Reduction of Source Term So Far as is Reasonably Practicable

A number of design features and decisions have been made for the BWRX-300 with the aim to mitigate source term SFAIRP. The list below is a summary:

- Elimination of carbon steels and replacement with stainless steels to mitigate source term resulting from FAC of the carbon steels
- Development of BWRX-300 specific chemistry regime, rather than adoption of generic BWR chemistry to best protect fuel cladding and structural steels
- Utilisation of high reliability modern fuel design to minimise fuel failures
- Utilisation of OLNCTM in combination with HWC to enable normal operation hydrogen concentrations to be below the trigger point for excessive N-16 production in the form of ammonia
- Used of DZO to minimise Co-60 buildup on reactor circuit system surfaces

- Inclusion of automatic isolated redundant passive cooling systems for in the event of accident or unplanned shutdown to contain radionuclides in the core

23.8.2 Primary Source Term – Realistic Model

ANSI/ANS-18.1-2020

The RM PST BWRX-300 RW and steam radiation concentrations are based on BWR operating data reported in the ANSI/ANS-18.1-2020 Standard (Reference 23-42). The standard provides bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR. The source term data is based on the cumulative industry experience at operating BWR plants, and measurements at several stations. It therefore reflects the influence of a number of observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs. Normal operation coolant source terms are obtained by applying the procedures of ANSI/ANS-18.1-2020 for estimation of typical source terms and adjusting the results, as appropriate, to assure conservative bases for design.

The ANSI/ANS-18.1-2020 standard is Revision 4 of the original American National Standard N237-1976 (ANS-18.1-1976), “Source Term Specification” prepared by the ANS-18.1 Working Group of the ANS. The ANSI/ANS-18.1-2016 BWR Reference Plant RW and steam concentrations were updated based on work undertaken by the EPRI reviewing the Technical Bases for Updating the ANSI/ANS 18.1-1999 Standard (Reference 23-44). This was based on data from operating domestic nuclear power plants collected from 1999 to 2015 and include contribution from normal operating events such as fuel leakers, implementation of different chemistry and mitigation strategies, and mid-cycle outages. A minor modification was subsequently made to the 2020 version of the ANSI/ANS 18.1 standard.

Radionuclide Classes

The radionuclides included in ANSI/ANS-18.1-2020 are categorised as FPs, APs, and activated CPs. The lists do not necessarily include all radionuclides that may be detectable or theoretically predicted to be present, but are focused on those significant to design, understanding of performance and that can be measured practically. The classes are defined as follows:

- Class 1 – Noble gas
- Class 2 – Halogens (iodines)
- Class 3 – Caesium, rubidium
- Class 4 – Water APs (N-16)
- Class 5 – Tritium
- Class 6 – Other nuclides

Except for Class 6, these classes are based on the grouping of elements with reasonably similar chemical and physical properties or similar behaviour during normal operation. Classes 1 to 3 account for volatile FPs and are those which RW and steam concentrations are most effected by fuel failures. Classes 4 and 5 concentrations are predominately based on activation. Class 6 contains all the activated CPs, non-volatile FPs and actinides. The reference radionuclide concentrations listed in ANSI/ANS-18.1-2020, are reproduced in Table 23-8.

Realistic Model PST Derivation

The determination of the BWRX-300 RM RW and reactor steam source terms uses the ANSI/ANS-18.1-2020 radionuclide concentrations listed in Table 23-8 adjusted using the

ANSI/ANS 18.1-2020 methodology to account for variations in key plant parameters, listed in Table 23-9.

While the BWRX-300 is an SMR, the fuel type, fuel burnup levels, fuel enrichment, and power to steam flow ratio are similar to the BWRs providing the operating data reviewed by EPRI (Reference 23-44) used to generate the ANSI/ANS-18.1-2020 standard. Based on the similarities, the ANSI/ANS-18.1-2020 standard is judged to be applicable to the BWRX-300.

BWR Reference Plant Adjustment Factors (AFs) are based on the following expression:

$$C = \frac{S}{M(\lambda+\beta)} \qquad \text{(Equation 1)}$$

where:

C = nuclide concentration (Bq/g),

S = nuclide generation rate (Bq/s),

M = RW mass (g),

λ = nuclide decay constant (s^{-1}),

β = the sum of all other removal rates of the radionuclide (s^{-1}).

The AFs are derived from the ratio of BWRX-300 to BWR Reference Plant concentrations shown in Equation 2.

$$AF = \frac{C_B}{C_R} = \frac{\frac{S_B}{M_B(\lambda+\beta_{i,B})}}{\frac{S_R}{M_R(\lambda+\beta_{i,R})}} = \frac{S_B M_R(\lambda+\beta_{i,R})}{S_R M_B(\lambda+\beta_{i,B})} \qquad \text{(Equation 2)}$$

Where the subscripts refer to:

B = BWRX-300 plant

R = BWR Reference Plant

i = removal rate for the six different radionuclide classes.

Given that the nuclide generation rate is assumed to scale in direct proportion to reactor thermal power, The AF ratio may be modified by substituting S for P:

$$AF = \frac{C_B}{C_R} = \frac{\frac{P_B}{M_B(\lambda+\beta_{i,B})}}{\frac{P_R}{M_R(\lambda+\beta_{i,R})}} = \frac{P_B M_R(\lambda+\beta_{i,R})}{P_R M_B(\lambda+\beta_{i,B})} \qquad \text{(Equation 3)}$$

From Equation 3, the radionuclide concentrations for the BWRX-300 simplifies to:

$$C_B = C_R \, . \, AF \qquad \text{(Equation 4)}$$

For each radionuclide class, the AF is calculated as discussed below.

Noble Gases (Radionuclide Class 1)

The BWRX-300 normal operation noble gas source term consists of the thirteen-principle noble gas FPs as observed in steam flowing from the reactor vessel which are provided in ANSI/ANS-18.1-2020. Concentrations in the RW are considered negligible under normal power operation because all the gases released to the coolant are assumed to be rapidly transported out of the vessel with the steam and removed from the system with the other non-condensable gases in the main condenser. Due to the rapid removal of the gases, the expected relative mix of gases does not depend on the reactor design. The RM noble gas reactor steam source term is listed in Table 23-10.

Fission Products and Activated Corrosion Products (Radionuclide Class 2, Class 3, and Class 6)

For Radionuclide Class 2, 3, and 6 radionuclides, the ANSI/ANS-18.1-2020 standard derives the radionuclide removal term, b, from the dual cleanup of steam condensate by the filter and demineralisers in the CFD and the RW by the filter-demineralizers in the CUW. The removal rate, b2,3,6 for nuclide Radionuclide Classes 2, 3, and 6, is given by:

$$\beta_{2,3,6} = \frac{(FA \times NA)+(NC \times FS \times NS \times NB)}{M} \qquad \text{(Equation 5)}$$

Where:

FA = CUW flow rate (g/s),

NA = fraction of nuclide removed in CUW (-),

NC = ratio of CFD system flow rate to steam flow rate (-),

FS = steam flow rate (g/s),

NS = steam carryover fraction (-),

NB = fraction of nuclide removed by the CFD system (-),

M = mass of water in the reactor vessel (g).

With values for all parameters listed in Table 9.

It is noted that the ANSI/ANS-18.1-2020 BWR Reference Plant configuration, is based on separate CUW and CFD cleanup terms. To account for the combined cleanup function of the CFD in the BWRX-300, Equation 5 is modified to represent the cleanup flows as follows:

$$\beta_{2,3,6} = \frac{(FA+FS \times NS \times NC) \times NB}{M} \qquad \text{(Equation 6).}$$

Class 2,3, and 6 radionuclides concentrations are adjusted based on Equations 6 and 4. The RM RW and reactor steam values are listed in Table 23-11, Table 23-12, Table 23-13, and Table 23-14.

Nitrogen-16 (Radionuclide Class 4)

The coolant AP of primary importance in BWRs is N-16. Based on the ANSI/ANS-18.1-2020 BWR Reference Plant and the BWRX-300 operating parameters, the N-16 levels of in steam leaving the reactor vessel for the plant without HWC is 1.85 MBq/gm (50 µCi/gm), and {9.25 MBq/gm (250 µCi/gm)} with high HWC injection rates. Under HWC conditions, the volatility of the N-16 in BWR water increases. As the FW hydrogen concentration is increased beyond approximately 0.4 ppm, the radiation levels in the MS increase rapidly, and then asymptotically approach levels 4 to 5 times the initial value at the higher FW hydrogen concentrations of approaching 2 ppm. The BWRX-300 will employ On-Line NobleChem™ technology along with the HWC which will allow the rate of hydrogen injection to be decreased such that the FW hydrogen concentration required is expected to be in the range of 0.2 to 0.4 ppm which implies no increase in operating dose rates due to increased concentrations of N-16 in the steam. The BWRX-300 HWC system maximum injection rate corresponds to a FW hydrogen concentration of 0.5 ppm with the plant at 100% power.

The normal operation source term is used to assess performance of radioactive waste management systems and other systems under normal operating conditions, the BWR Reference Plant concentrations of 2.2E+00 MBq/gm (6.0E+01 µCi/gm) for RW and 1.9E+00 MBq/gm (5.0E+01 µCi/gm) for reactor steam are conservative for the normal operation source term. The normal operating N-16 concentrations are presented in Table 23-15.

Tritium (Radionuclide Class 5)

The BWRX-300 does not use boron in RW, so the production of tritium is limited to activation of naturally occurring deuterium in the primary coolant and, to a lesser extent, as a fission product in the fuel. In addition, the BWRX-300 design is capable of recycling all wastewater. The recycling of RW allows the BWRX-300 to operate with a zero liquid release strategy for radiological effluents that results in the release of tritium as a gaseous effluent instead of a liquid effluent. Zero liquid release for radiological effluents is a philosophy of processing wastewater in a way such that no wastewater generated at the power plant is released into local bodies of water and the environment in general.

Because tritium has a long half-life (12 years) and is not affected by cleanup processes in the system, the concentration of tritium in reactor coolant is controlled by the rate of loss of water from the system by evaporation or leakage. The concentration of tritium is influenced by factors such as evaporation rates from fuel pool water and the addition of pure water. RM RW and reactor steam source term values are given in Table 23-16.

23.8.3 Primary Source Term – Design Basis

Design Basis PST Derivation

The DB coolant source term employs a GEH fuel clad defect model that applies the following fuel leakage to the normal operation coolant source term which bound the expected fuel leakage rates that will be observed in an operating BWRX-300 reactor:

- Noble gas fuel leak rate of {740 MBq/sec (20,000 µCi/sec)} after a 30-minute decay
- I-131 fuel leak rate of {9.25 MBq/sec (250 µCi/sec)}

The GEH fuel clad defect model is developed based on more than 50 years of BWR fuel leakage operating data from several representative BWRs that experienced a variety of fuel defects. This review of modern fuel performance operating data identified two trends. First, the reliability of BWR fuel has increased dramatically over the last 25 years driven by improvements in fuel fabrication, design, and plant operation, and chemistry while at the same time techniques for managing leaking fuel have been modernized. Second, BWR operators that utilize modern fuel defect detection, characterization, and suppression techniques do not observe fuel leakage rates as high as those established the BWRX-300 clad defect model.

The GEH fuel clad defect model assumes the operator of any BWRX-300 will employ up to date modern BWR fuel failure detection, characterization, and suppression procedures similar to those documented in the “Global Nuclear Fuel (GNF) Fuel Operating Guidelines,” (Reference 23-45) or in EPRI) Boiling Water Reactor Fuel Reliability Monitoring and Failure Evaluation Handbook (Reference 23-46). If these procedures, or similar techniques, are employed by a BWRX-300 operator the GEH clad defect model will bound fuel leakage rates that will be observed during power operation of the reactor. The key GEH clad defect model parameters are summarised in Table 23-17.

Noble Gases (Radionuclide Class 1)

The DB noble gas source terms are developed by normalizing the ANSI/ANS-18.1-2020 normal operation noble gas source term mixture to a fuel release rate of 3,219 MBq/sec (87,000 µCi/sec) at 30 minutes decay. The DB noble gas radionuclide source term rate after 30-minute decay has been used as a conventional measure of the fuel leakage rate, since it is conveniently measurable and was consistent with the nominal 30-minute offgas holdup system used on a number of early plants.

NUREG-0800, "U.S. Nuclear Regulatory Commission (USNRC) Standard Review Plan, Branch Technical Position (BTP), Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," (Reference 23-47), states that releases should be based on 3.7 MBq/sec (100 µCi/sec) per MWt with 30 minutes of decay).

For the 870 MWt BWRX-300 this is a release rate of 3,219 MBq/s (87,000 µCi/sec), and the DB source term for noble gas is based on a release rate of {740 MBq/s (20,000 µCi/sec)} the BTP 11-5 AF is 4.35 is applied to the ANSI/ANS-18.1-2020 noble gas source term. The principal DB noble gas radionuclides source terms are presented in Table 23-18.

Iodine (Radionuclide Class 2)

The DB radioiodine concentrations are based on the relative mix of radioiodines in RW predicted by the data of ANSI/ANS-18.1-2020 with magnitudes increased by normalizing the I-131 concentrations to the GEH fuel clad defect model I-131 release rate of {9.25 MBq/sec (250 µCi/sec)}. ANSI/ANS-18.1-2020 specifies expected concentrations of the 5 principal radioiodines in RW for a BWR Reference Plant design and provides bases for adjusting the concentrations for plants with relevant plant parameters that do not match those of the Reference Plant. The DB radioiodine concentrations in steam and water are presented in Table 23-19.

The ratio of concentration in reactor steam to concentration in RW (carryover ratio) is based on the BWR GALE Code, NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors," (Reference 23-48) and taken to be 0.02 for radioiodines. Consequently, the DB radioiodine concentrations in steam are calculated by multiplying the DB radioiodine water concentrations in Table 11.1-5 by the factor 0.02.

Other Fission Products (Radionuclide Class 3 and Class 6)

DB RW concentrations are determined by adjusting RM values (as derived in Table 23-12 (Class 3) and Table 23-14 (Class 6 FPs) by the same I-131 scaling factor discussed in Section 23.8.3. Reactor steam concentration is determined using RW DB values adjusted using a steam carryover fraction of 10-3 (Table 23-9). The DB volatile and non-volatile FPs RW and reactor steam source terms are given in Table 23-20 (Class 3 FPs) and Table 23-21 (Class 6 FPs).

Activated Corrosion Products (Class 6 Radionuclides)

The BWRX-300 DB concentrations of the activated CPs are not normalised using the fuel leakage model because they do not originate in the fuel but are produced by corrosion of irradiated system materials in the reactor and associated piping.

Nitrogen-16 (Class 4 Radionuclide)

The BWRX-300 DB N-16 coolant concentrations are used for shielding design assessments and the HWC injection system maximum injection rate can result in FW hydrogen concentration of 0.5 ppm an increase in N-16 concentration in steam by a factor of 2.5 is possible at the maximum HWC injection rate. Thus, the BWRX-300 DB N-16 concentrations are conservatively estimated to be 2.2E+00 MBq/gm (6.0E+01 µCi/gm) for RW and

4.6E+00 MBq/gm (1.3E+02 μCi/gm) for reactor steam. The DB N-16 concentrations in steam and RW are shown in Table 23-22.

Tritium (Class 5 Radionuclide)

Plant process water and steam have a common tritium concentration. ANSI/ANS-18.1-2020 specifies a concentration of 518 Bq/gm (0.014 μCi/gm) in both RW and reactor steam based on BWR operating data. Under the zero liquid release strategy, tritium levels in the coolant are conservatively assumed to reach an equilibrium level of twice that of a typical BWR that discharges liquid effluents.

Thus, for the BWRX-300, the tritium concentration in RW and steam is 1036 Bq/gm (0.028 μCi/gm).

There may be circumstances in which release is required. These releases can be processed to meet specification for release and these events would be planned, controlled, and monitored. permit. Tritium RW and steam concentrations are listed in Table 23-23.

There may be circumstances in which release is required, such as in-leakage from either ground water or radwaste water, these releases can be processed to meet specification for release and these events would be planned, controlled, and undertaken under permit.

Argon-41

Argon-41 is produced in the reactor coolant as a consequence of neutron activation of naturally occurring Argon-40 in air that is entrained in the FW. The Argon-41 gas is carried out of the vessel with the steam and stripped from the system with the non-condensable gases in the main condenser. Observed Argon-41 levels are highly variable due to the variability in air in-leakage rates into the system. NUREG-0016 (Reference 23-48) specifies a RM Argon-41 release rate from the vessel into the offgas treatment system of 40 μCi/sec. This value is considered conservative as it bounds the available experimental database; this value is provided in Table 23-17.

23.8.4 Primary Source Term - Validation

UK ABWR Regulatory Credit

BWRs have prior credibility in the UK, where the UK ABWR received Design Acceptance Confirmation (DAC) (Reference 23-1) and Statement of Design Acceptability (SoDA) (Reference 23-2) in 2017 following a four step GDA process.

The UK ABWR source term was produced with major effort and was subject to significant regulatory scrutiny, as the initial submission was deemed inadequate requiring resolution in response to a Regulatory Issue (RI), RI-ABWR-0001 (Reference 23-49), being raised.

The final submission was deemed adequate in scope and approach, where the regulator noted the following in its conclusions when closing out the RI:

- *The use of relevant plant operating experience, utilising the broadest data set that is considered pertinent, gives confidence in the defined values. Where suitable data does not exists recourse is made to other methods in an satisfactory manner*
- *While the responses have been updated several times throughout my assessment, sufficient evidence has been documented to capture and understand the basis of the UK ABWR source terms should this need to be revisited in the future*

There were numerous other conclusions drawn, but these two are the most significant. Given the prior endorsement of the UK ABWR source term derivation and justification and the suitability of the data presented and quality of supporting documentation, a comparison of BWRX-300 PST and UK ABWR PST is provided below as suitable validation.

PST - BWRX-300 Comparison with UK ABWR

While the BWRX-300 will adopt an almost identical chemistry regime to that proposed for the UK ABWR, it is recognised that there are some design differences that would be expected to affect source term. Most notable of these are the configurations of the:

- FW heater drains (cascaded for BWRX-300 vs forward pumped for UK ABWR)
- cleanup systems (CFD only for BWRX-300 vs CFD and CUW for UK ABWR)

However, since both design difference can be accounted for in the AF methodology presented in ANSI/ANS-18.1-2020, these can be disregarded in the following analysis.

It is noted that the UK ABWR PST contained 100 radionuclides, based on measured OPEX and on calculations to fill in gaps. The ANSI/ANS-18.1-2020 standard and, therefore, the BWRX-300 PST contains 65 radionuclide all of which are directly measured in ten BWR plants in the USA. Comparison is made where there is radionuclide overlap in the two datasets.

Matching radionuclides derived in Section 23.8.2 (RM) and Section 23.8.3 (DB) for the BWRX-300 have been compared with the same radionuclides derived for the UK ABWR Primary Source Term (PST) (Reference 23-50). To make a meaningful comparison both the RM and DB PST for the UK ABWR have been adjusted using the methodology detailed in ANSI/ANS-18.1-2020, using appropriate UK ABWR plant `parameters. In addition, the UK ABWR RM PST has been modified for equivalent steam carryover fractions to enable a like-for-like comparison; BWRX-300 uses the same steam carryover fractions for RM and DB, whereas UK ABWR used different RM and DB values for Class 3 and 6 radionuclides.

Rather than directly compare individual radionuclides, comparison is made at the radionuclide class level to understand trends, broad equivalence, conservatism or otherwise. For a particular radionuclide class (for example Class 1 – noble gases), the concentrations, C, of the n radionuclides are summed and the product raised to the power 1/n to obtain the geometric mean of the population, as shown in Equation 7:

$$\prod_n C^{1/n} \qquad \text{(Equation 7)}$$

The comparison is obtained from the ratio of geometric means of the BWRX-300:UK ABWR for each radionuclide class.

Comparison by geometric mean is selected as there is significant order of magnitude variations for radionuclides within a given radionuclide class. Using a geometric mean does not skew the calculation of the mean value to the larger individual value. In addition, the geometric mean is used for comparison due to the fluctuation on a nuclide-by-nuclide basis as the underpinning OPEX are completely different. This is particularly of note for the RM comparison, where the UK ABWR is based on OPEX containing no fuel failures, but where the ANSI/ANS-18.1-2020 reference plant OPEX contains some plant data where fuel failures have occurred.

Due to the difference in inclusion of fuel failures a meaningful comparison would expect to see:

- Radionuclide Classes 1, 2 and 3 be relatively high and conservative due to the nuclides within being volatile FPs that can escape the fuel pin following failure
- Radionuclide Classes 4, 5 and 6 are expected to be broadly comparable as their concentrations are less effected by fuel failures, where the nuclides within are APs, activated CPs, non-volatile FPs and actinides

Realistic Model Comparison

The RM comparison data presented in Table 23-24 demonstrates the expected trend between the volatile and non-volatile radionuclide classes. Radionuclide Class 1 steam is noted as being highly conservative relative to the other radionuclide classes in the volatile FP group. The conservatism is heavily influenced by K-85 and Xe-131m. The difference is a function of the way they are calculated for each reactor, effecting the nuclides that are both long lived and in low concentrations in the system most notably. The concentration in both assessments is low, but the ANSI-ANS-18.1 data is crucially more conservative for design.

Design Basis Comparison

The DB comparison, in Table 23-25, shows that the GEH fuel clad defect model (based on OPEX) is more closely aligned to the UK ABWR data also based on different OPEX of fuel failures.

The comparison results indicate that the GEH source term derivation of source term values is broadly similar to that developed previously for UK ABWR and, therefore suitable for use in Step 2 GDA.

Table 23-1: Reactor Water Quality Design (>10% Power) and Maximum Values

Water Quality Parameter	Values	
	Design Values	Maximum Values
pH	-	4-10
Conductivity @25°C	≤0.15 μS/cm (with Zn injection) ≤0.08 μS/cm (without Zn injection)	5 μS/cm
Chloride	≤1 ppb	100 ppb
Sulfate	≤2 ppb	100 ppb
Soluble Co-60	≤10^{-4} μCi/ml (≤ 3.7 Bq/ml)	-
Soluble Copper	≤1 ppb	-
Silica	≤300 ppb	-
Dissolved Oxygen	40-100 ppb	8 ppm

Note:

006N6766, (Reference 23-32)

Table 23-2: Condensate Filter and Demineraliser Aystem Inlet Water Quality Design (>10% Power) and Maximum Values

Water Quality Parameter	Values	
	Design Values	Maximum Values
Conductivity @25°C	≤0.1 µS/cm	2.0 µS/cm
Chloride	≤5 ppb	100 ppb
Sulfate	≤5 ppb	100 ppb
Dissolved Oxygen	-	10-200 ppb

Note:

006N6766, (Reference 23-32)

Table 23-3: Feedwater Quality Design (> 10% Power) and Maximum Values

Water Quality Parameter	Values	
	Design Values	Maximum Values
Conductivity @25^{o}C	≤0.06 μS/cm	2.0 μS/cm
Total Iron	≤ 1.0 ppb	
Total Copper	≤ 0.05 ppb	
Chloride	-	100 ppb
Sulfate	-	100 ppb

Note:

006N6766, (Reference 23-32)

Table 23-4: Reactor Water Quality During Operations > 10%, and < 10% Power, but >93°C

Water Quality Parameter	Design Values	
	Above 10% Power	Below 10% Power (>93°C)
Conductivity @25°C	≤0.15 μS/cm (with Zn injection) ≤0.08 μS/cm (without Zn injection)	≤0.3 μS/cm
Chloride	≤1 ppb	≤5 ppb
Sulfate	≤2 ppb	≤5 ppb
Soluble Co-60	≤10^{-4} μCi/ml (≤ 3.7 Bq/ml)	-
Soluble Copper	≤1 ppb	-
Silica	≤300 ppb	-

Note:

006N6766, (Reference 23-32)

Table 23-5: Fuel Pool Cooling and Cleanup Water Quality Normal and Maximum Values

Water Quality Parameter	Values	
	Design Values	Maximum Values
pH	-	4-10
Conductivity @25°C	≤1.3 μS/cm	5 μS/cm
Chloride	≤100 ppb	100 ppb
Sulfate	≤100 ppb	100 ppb
Total Metals	≤ 150 ppb	-
Silica	≤300 ppb	-
Total Organic Carbon	≤400 ppb	
Dissolved Oxygen	-	8 ppm

Note:

006N6766, (Reference 23-32)

Table 23-6: Water Quality for Demineraliser Makeup Water for Tanks

Water Quality Parameter	Design Value
Conductivity @25°C	< 0.06 μS/cm
Chloride	< 1ppb
Sulfate	< 1ppb
Post UV Chloride and Sulfate	< 2ppb
Total Organic Carbon	≤ 30ppb
Silica	≤ 100 ppb

Note

006N6766, (Reference 23-32)

Table 23-7: **Isolation Condenser Pools Cooling and Cleanup System Design Values**

Water Quality Parameters	Design Values
Conductivity at 25°C (77°F)	≤ 5 μS/cm
Chloride	≤ 200 ppb
Sulfate	≤ 200 ppb
TOC	≤ 1000 ppb

Note:

006N6766, (Reference 23-32)

Table 23-8: Reference BWR Radionuclide Concentrations in Reactor Water and Steam, Taken from ANSI/ANS-18.1-2020

Radionuclide	Reactor Water	Reactor Steam	Reactor Water	Reactor Steam
	μCi/g	μCi/g	MBq/g	MBq/g
Class 1				
Kr-83m		5.90E-04		2.20E-05
Kr-85m		1.60E-05		5.90E-07
Kr-85		4.00E-06		1.50E-07
Kr-87		1.10E-04		4.10E-06
Kr-88		6.20E-05		2.30E-06
Kr-89		2.10E-02		7.80E-04
Xe-131m		3.30E-06		1.20E-07
Xe-133m		2.00E-06		7.40E-08
Xe-133		3.00E-05		1.10E-06
Xe-135m		5.40E-04		2.00E-05
Xe-135		3.00E-04		1.10E-05
Xe-137		9.90E-04		3.70E-05
Xe-138		2.80E-03		1.00E-04
Class 2				
I-131	8.30E-05	1.70E-06	3.07E+00	6.29E-02
I-132	1.20E-03	2.30E-05	4.44E+01	8.51E-01
I-133	6.50E-04	1.30E-05	2.41E+01	4.81E-01
I-134	4.50E-03	9.00E-05	1.67E+02	3.33E+00
I-135	1.40E-03	2.70E-05	5.18E+01	9.99E-01
Class 3				
Rb-89	5.00E-03	5.00E-06	1.85E+02	1.85E-01
Cs-134	3.00E-05	3.00E-08	1.11E+00	1.11E-03
Cs-136	2.40E-05	2.40E-08	8.88E-01	8.88E-04
Cs-137	4.60E-05	4.60E-08	1.70E+00	1.70E-03
Cs-138	4.60E-05	4.60E-08	1.70E+00	1.70E-03
Ba-137m	5.20E-03	5.20E-06	1.92E+02	1.92E-01
Class 4				
N-16	6.00E+01	5.00E+01	2.20E+00	1.90E+00
Class 5				
H-3	1.40E-02	1.40E-02	5.20E-04	5.20E-04

Radionuclide	Reactor Water	Reactor Steam	Reactor Water	Reactor Steam
	μCi/g	μCi/g	MBq/g	MBq/g
Class 6				
Na-24	1.20E-03	1.20E-06	4.44E+01	4.44E-02
P-32	4.00E-05	4.00E-08	1.48E+00	1.48E-03
Cr-51	9.70E-04	9.70E-07	3.59E+01	3.59E-02
Mn-54	4.80E-04	4.80E-07	1.70E+01	1.70E-02
Mn-56	3.30E-03	3.30E-06	1.22E+02	1.22E-01
Fe-55	1.00E-03	1.00E-06	3.70E+01	3.70E-02
Fe-59	2.60E-04	2.60E-07	9.62E+00	9.62E-03
Co-58	2.30E-04	2.30E-07	8.51E+00	8.51E-03
Co-60	4.40E-04	4.40E-07	1.63E+01	1.63E-02
Ni-63	1.00E-06	1.00E-09	3.70E-02	3.70E-05
Cu-64	5.90E-03	5.90E-06	2.18E+02	2.18E-01
Zn-65	2.10E-04	2.10E-07	7.77E+00	7.77E-03
Sr-89	3.00E-06	3.00E-09	1.11E-01	1.11E-04
Sr-90	1.50E-07	1.50E-10	5.55E-03	5.55E-06
Y-90	1.50E-07	1.50E-10	5.55E-03	5.55E-06
Sr-91	2.40E-03	2.40E-06	8.88E+01	8.88E-02
Sr-92	7.00E-03	7.00E-06	2.59E+02	2.59E-01
Y-91	4.00E-05	4.00E-08	1.48E+00	1.48E-03
Y-92	2.00E-03	2.00E-06	7.40E+01	7.40E-02
Y-93	1.70E-04	1.70E-07	6.29E+00	6.29E-03
Zr-95	8.30E-05	8.30E-08	3.07E+00	3.07E-03
Nb-95	8.30E-05	8.30E-08	3.07E+00	3.07E-03
Mo-99	4.10E-04	4.10E-07	1.52E+01	1.52E-02
Tc-99m	4.10E-04	4.10E-07	1.52E+01	1.52E-02
Ru-103	2.00E-05	2.00E-08	7.40E-01	7.40E-04
Rh-103m	2.00E-05	2.00E-08	7.40E-01	7.40E-04
Ru-106	3.00E-06	3.00E-09	1.11E-01	1.11E-04
Rh-106	3.00E-06	3.00E-09	1.11E-01	1.11E-04
Ag-110m	1.00E-06	1.00E-09	3.70E-02	3.70E-05
Te-129m	4.00E-05	4.00E-08	1.48E+00	1.48E-03
Te-131m	5.80E-05	5.80E-08	2.15E+00	2.15E-03
Te-132	1.00E-05	1.00E-08	3.70E-01	3.70E-04
Ba-140	4.20E-04	4.20E-07	1.55E+01	1.55E-02
La-140	4.20E-04	4.20E-07	1.55E+01	1.55E-02

Radionuclide	**Reactor Water**	**Reactor Steam**	**Reactor Water**	**Reactor Steam**
	μCi/g	**μCi/g**	**MBq/g**	**MBq/g**
Ce-141	2.00E-05	2.00E-08	7.40E-01	7.40E-04
Ce-144	3.00E-06	3.00E-09	1.11E-01	1.11E-04
Pr-144	3.00E-06	3.00E-09	1.11E-01	1.11E-04
W-187	2.60E-04	2.60E-07	9.62E+00	9.62E-03
Np-239	3.40E-04	3.40E-07	1.26E+01	1.26E-02

Note:

“Radioactive Source Term for Normal Operation of Light Water Reactors,” (Reference 23-42)

Table 23-8: Plant Parameters for Source Term Adjustment

Parameter	Symbol	Units	Reference Plant	BWRX-300
Reactor Thermal Power	*P*	MW_t	3400	870
Reactor Water Mass	*M*	g	1.70E+08	1.47E+08
CUW Flow Rate	*NC*	g/s	1.61E+04	5.06E+03
Steam Flow Rate	*FS*	g/s	1.90E+06	5.02E+05
Ratio of Condensate Demineralizer Flow Rate to Steam Flow Rate	*NC*	(-)	1	1
Removal Fraction by CUW	*NA*	(-)		
Radionuclide Class 1			0	N/A
Radionuclide Class 2			0.9	N/A
Radionuclide Class 3			0.5	N/A
Radionuclide Class 4			1	N/A
Radionuclide Class 5			0	N/A
Radionuclide Class 6			0.9	N/A
Removal Fraction by CFD	*NB*	(-)		
Radionuclide Class 1			0	0
Radionuclide Class 2			0.9	0.9
Radionuclide Class 3			0.5	0.5
Radionuclide Class 4			1	1
Radionuclide Class 5			0	0
Radionuclide Class 6			0.9	0.9
Steam Carryover Fraction	*NS*	(-)		
Radionuclide Class 1			1	1
Radionuclide Class 2			2.0E-02	2.0E-02
Radionuclide Class 3			1.0E-03	1.0E-03
Radionuclide Class 4			1	1
Radionuclide Class 5			1	1
Radionuclide Class 6			1.0E-03	1.0E-03

Table 23-9: Realistic Model Noble Gas Reactor Steam Source Term

Isotope	Decay Constant (Per hour)	Steam Concentration (MBq/g)	Steam Concentration (µCi/g)
Kr-83m	3.79E-01	2.2E-05	5.9E-04
Kr-85m	1.55E-01	5.9E-07	1.6E-05
Kr-85	7.35E-06	1.5E-07	4.0E-06
Kr-87	5.45E-01	4.1E-06	1.1E-04
Kr-88	2.44E-01	2.3E-06	6.2E-05
Kr-89	1.32E+01	7.8E-04	2.1E-02
Xe-131m	2.44E-03	1.2E-07	3.3E-06
Xe-133m	1.32E-02	7.4E-08	2.0E-06
Xe-133	5.51E-03	1.1E-06	3.0E-05
Xe-135m	2.72E+00	2.0E-05	5.4E-04
Xe-135	7.58E-02	1.1E-05	3.0E-04
Xe-137	1.09E+01	3.7E-05	9.9E-04
Xe-138	2.95E+00	1.0E-04	2.8E-03

Table 23-10: Realistic Model Iodine Reactor Water/Steam Source Terms

Isotope	Decay Constant	Water Concentration		Steam Concentration	
	(per hour)	(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
I-131	3.60E-03	2.7E-06	7.4E-05	5.6E-08	1.5E-06
I-132	3.02E-01	2.7E-05	7.4E-04	5.2E-07	1.4E-05
I-133	3.33E-02	2.0E-05	5.5E-04	4.1E-07	1.1E-05
I-134	7.92E-01	7.9E-05	2.1E-03	1.6E-06	4.3E-05
I-135	1.06E-01	3.9E-05	1.1E-03	7.5E-07	2.0E-05

Table 23-11: Realistic Model Volatile Fission Products Reactor Water/Steam Source Terms

Isotope*	Decay Constant	Water Concentration		Steam Concentration	
	(per hour)	(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
Rb-89	2.75E+00	5.7E-05	1.5E-03	5.7E-08	1.5E-06
Cs-134	3.83E-05	8.3E-07	2.2E-05	8.3E-10	2.2E-08
Cs-136	2.20E-03	6.5E-07	1.8E-05	6.5E-10	1.8E-08
Cs-137/Ba-137m	2.62E-06	1.3E-06	3.4E-05	1.3E-09	3.4E-08
Cs-138	1.25E+00	6.2E-05	1.7E-03	6.2E-08	1.7E-06

Note:

*Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

Table 23-12: Realistic Model Activated Corrosion Products Reactor Water/Steam Source Terms

Isotope	Decay Constant	Water Concentrations*		Steam Concentrations*	
	(per hour)	(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
Na-24	4.63E-02	3.0E-05	8.2E-04	3.0E-08	8.2E-07
P-32	2.03E-03	1.2E-06	3.3E-05	1.2E-09	3.3E-08
Cr-51	1.04E-03	3.0E-05	8.0E-04	3.0E-08	8.0E-07
Mn-54	9.25E-05	1.5E-05	4.0E-04	1.5E-08	4.0E-07
Mn-56	2.69E-01	5.7E-05	1.5E-03	5.7E-08	1.5E-06
Fe-55	2.89E-05	3.1E-05	8.3E-04	3.1E-08	8.3E-07
Fe-59	6.49E-04	8.0E-06	2.2E-04	8.0E-09	2.2E-07
Co-58	4.08E-04	7.1E-06	1.9E-04	7.1E-09	1.9E-07
Co-60	1.50E-05	1.4E-05	3.7E-04	1.4E-08	3.7E-07
Ni-63	7.90E-07	3.1E-08	8.3E-07	3.1E-11	8.3E-10
Cu-64	5.46E-02	1.5E-04	3.9E-03	1.5E-07	3.9E-06
Zn-65	1.18E-04	6.5E-06	1.7E-04	6.5E-09	1.7E-07
Ag-110m	1.16E-04	3.1E-08	8.3E-07	3.1E-11	8.3E-10
W-187	2.92E-02	7.0E-06	1.9E-04	7.0E-09	1.9E-07

Note:

* Normal operational concentrations are the same as the DB concentrations.

Table 23-13: Realistic Model Non-Volatile Fission Products Reactor Water/Steam Source Terms

Isotope*	Decay Constant	Water Concentrations*		Steam Concentrations*	
	(per hour)	(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
Sr-89	5.72E-04	9.2E-08	2.5E-06	9.2E-11	2.5E-09
Sr-90	2.75E-06	4.6E-09	1.2E-07	4.6E-12	1.2E-10
Y-90	2.75E-06	4.6E-09	1.2E-07	4.6E-12	1.2E-10
Sr-91	7.20E-02	5.6E-05	1.5E-03	5.6E-08	1.5E-06
Sr-92	2.61E-01	1.2E-04	3.3E-03	1.2E-07	3.3E-06
Y-91	4.94E-04	1.2E-06	3.3E-05	1.2E-09	3.3E-08
Y-92	1.96E-01	3.7E-05	1.0E-03	3.7E-08	1.0E-06
Y-93	6.81E-02	4.0E-06	1.1E-04	4.0E-09	1.1E-07
Zr-95/Nb-95	4.51E-04	2.5E-06	6.9E-05	2.5E-09	6.9E-08
Mo-99/Tc-99m	1.05E-02	1.2E-05	3.2E-04	1.2E-08	3.2E-07
Ru-103/Rh-103m	7.36E-04	6.1E-07	1.7E-05	6.1E-10	1.7E-08
Ru-106/Rh-106	7.73E-05	9.2E-08	2.5E-06	9.2E-11	2.5E-09
Te-129m	8.60E-04	1.2E-06	3.3E-05	1.2E-09	3.3E-08
Te-131m	2.31E-02	1.6E-06	4.3E-05	1.6E-09	4.3E-08
Te-132	9.01E-03	2.9E-07	7.9E-06	2.9E-10	7.9E-09
Ba-140/La-140	2.27E-03	1.3E-05	3.5E-04	1.3E-08	3.5E-07
Ce-141	8.88E-04	6.1E-07	1.7E-05	6.1E-10	1.7E-08
Ce-144/Pr-144	1.01E-04	9.2E-08	2.5E-06	9.2E-11	2.5E-09
Np-239	1.23E-02	9.8E-06	2.7E-04	9.8E-09	2.7E-07

Note:

*Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

Table 23-14: Realistic Model N-16 Reactor Water/Steam Source Terms

Isotope	Half-Life (sec)	Water Concentration		Steam Concentration*	
		(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
N-16	7.1	2.2E+00	6.0E+01	1.9E+00	5.0E+01

Note:

* Valid at core exit.

Table 23-15: Realistic Model H-3 Reactor Water/Steam Source Terms

Isotope	Half-Life (years)	Water Concentration		Steam Concentration	
		(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
H-3	12.3	5.2E-04	1.4E-02	5.2E-04	1.4E-02

Table 23-16: Source Term Design Basis Parameters

Parameter	Value
Total of the DB release rates of the 13 noble gases (30-minute decay reference, t=30)	3219 MBq/s (87,000 µCi/s)
Normal operational noble gas release rate	740 MBq/s (20,000 µCi/s)
DB I-131 radioiodine core release rate	9.25 MBq/s (250 µCi/s)
Normal operational Ar-41 release rate into the offgas treatment system	1.5 MBq/s (40 µCi/s)

Table 23-17: Design Basis Noble Gas Reactor Steam Source Term

Isotope	Decay Constant (Per hour)	Steam Concentration		Source Term at t=30min	
		(MBq/g)	(μCi/g)	(MBq/s)	(μCi/s)
Kr-83m	3.79E-01	4.9E-04	1.3E-02	8.8E+02	2.4E+04
Kr-85m	1.55E-01	1.3E-05	3.6E-04	2.7E+01	7.2E+02
Kr-85	7.35E-06	3.3E-06	8.9E-05	7.2E+00	2.0E+02
Kr-87	5.45E-01	9.1E-05	2.5E-03	1.5E+02	4.1E+03
Kr-88	2.44E-01	5.1E-05	1.4E-03	9.9E+01	2.7E+03
Kr-89	1.32E+01	1.7E-02	4.7E-01	5.2E+01	1.4E+03
Xe-131m	2.44E-03	2.7E-06	7.4E-05	6.0E+00	1.6E+02
Xe-133m	1.32E-02	1.7E-06	4.5E-05	3.6E+00	9.7E+01
Xe-133	5.51E-03	2.5E-05	6.7E-04	5.4E+01	1.5E+03
Xe-135m	2.72E+00	4.5E-04	1.2E-02	2.5E+02	6.8E+03
Xe-135	7.58E-02	2.5E-04	6.7E-03	5.2E+02	1.4E+04
Xe-137	1.09E+01	8.2E-04	2.2E-02	7.7E+00	2.1E+02
Xe-138	2.95E+00	2.3E-03	6.3E-02	1.2E+03	3.1E+04

Table 23-18: Design Basis Iodine Reactor Water/Steam Source Terms

Isotope	Decay Constant (per hour)	Water Concentration		Steam Concentration	
		(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
I-131	3.60E-03	6.7E-04	1.8E-02	1.4E-05	3.7E-04
I-132	3.02E-01	6.7E-03	1.8E-01	1.3E-04	3.5E-03
I-133	3.33E-02	5.0E-03	1.3E-01	1.0E-04	2.7E-03
I-134	7.92E-01	1.9E-02	5.3E-01	3.9E-04	1.1E-02
I-135	1.06E-01	9.6E-03	2.6E-01	1.8E-04	5.0E-03

Table 23-19: Design Basis Volatile Fission Products Reactor Water/Steam Source Terms

Isotope*	Decay Constant	Water Concentration		Steam Concentration	
	(per hour)	(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
Rb-89	2.75E+00	1.4E-02	3.8E-01	1.4E-05	3.8E-04
Cs-134	3.83E-05	2.0E-04	5.5E-03	2.0E-07	5.5E-06
Cs-136	2.20E-03	1.6E-04	4.3E-03	1.6E-07	4.3E-06
Cs-137/Ba-137m	2.62E-06	3.1E-04	8.4E-03	3.1E-07	8.4E-06
Cs-138	1.25E+00	1.5E-02	4.1E-01	1.5E-05	4.1E-04

Note:

*Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

Table 23-20: Design Basis Non-Volatile Fission Products Reactor Water/Steam Source Terms

Isotope*	Decay Constant	Water Concentration		Steam Concentration	
	(per hour)	(MBq/g)	(μCi/g)	(MBq/g)	(μCi/g)
Sr-89	5.72E-04	2.3E-05	6.1E-04	2.3E-08	6.1E-07
Sr-90	2.75E-06	1.1E-06	3.1E-05	1.1E-09	3.1E-08
Y-90	2.75E-06	1.1E-06	3.1E-05	1.1E-09	3.1E-08
Sr-91	7.20E-02	1.4E-02	3.7E-01	1.4E-05	3.7E-04
Sr-92	2.61E-01	3.0E-02	8.0E-01	3.0E-05	8.0E-04
Y-91	4.94E-04	3.0E-04	8.1E-03	3.0E-07	8.1E-06
Y-92	1.96E-01	9.1E-03	2.5E-01	9.1E-06	2.5E-04
Y-93	6.81E-02	9.9E-04	2.7E-02	9.9E-07	2.7E-05
Zr-95/Nb-95	4.51E-04	6.2E-04	1.7E-02	6.2E-07	1.7E-05
Mo-99/Tc-99m	1.05E-02	2.9E-03	7.9E-02	2.9E-06	7.9E-05
Ru-103/Rh-103m	7.36E-04	1.5E-04	4.1E-03	1.5E-07	4.1E-06
Ru-106/Rh-106	7.73E-05	2.3E-05	6.1E-04	2.3E-08	6.1E-07
Te-129m	8.60E-04	3.0E-04	8.1E-03	3.0E-07	8.1E-06
Te-131m	2.31E-02	3.9E-04	1.1E-02	3.9E-07	1.1E-05
Te-132	9.01E-03	7.2E-05	1.9E-03	7.2E-08	1.9E-06
Ba-140/La-140	2.27E-03	3.1E-03	8.5E-02	3.1E-06	8.5E-05
Ce-141	8.88E-04	1.5E-04	4.1E-03	1.5E-07	4.1E-06
Ce-144/Pr-144	1.01E-04	2.3E-05	6.1E-04	2.3E-08	6.1E-07
Np-239	1.23E-02	2.4E-03	6.5E-02	2.4E-06	6.5E-05

Note:

*Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

Table 23-21: Design Basis N-16 Reactor Water/Steam Source Terms

Isotope	Half-Life (sec)	Water Concentration		Steam Concentration*	
		(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
N-16	7.1	2.2E+00	6.0E+01	4.6E+00	1.3E+02

Note:

* Valid at core exit.

Table 23-22: Design Basis H-3 Reactor Water/Steam Source Terms

Isotope	Half-Life (years)	Water Concentration		Steam Concentration	
		(MBq/g)	(µCi/g)	(MBq/g)	(µCi/g)
H-3	12.3	1.0E-03	2.8E-02	1.0E-03	2.8E-02

Table 23-23: Geometric Mean of BWRX-300:UK ABWR by Radionuclide Class for Realistic Model Source Term

PST	RN Class 1	RN Class 2	RN Class 3	RN Class 4	RN Class 5	RN Class 6
Water	N/A	26.36	15.49	1.47	2.93	0.71
Steam	245.41	26.35	15.49	3.07	2.93	0.71

Notes:

- The UK ABWR ‘Best Estimate’ PST has been used as the basis for comparison with the BWRX-300 ‘RM’ as they are equivalent assessments.
- The UK ABWR ‘Best Estimate’ PST has been adjusted using the methodology detailed in ANSI/ANS 18.1-2020 to allow for comparison.
- Only radionuclides that are included in both the UK ABWR PST and in ANSI/ANS-18.1-2020 have been included in the comparison. 12 nuclides in Radionuclide Class 1, 5 nuclides in Radionuclide Class 2, 4 nuclides in Radionuclide Class 3, 1 nuclide in Radionuclide Class 4, 1 nuclide in Radionuclide Class 5 and 15 nuclides in Radionuclide Class 6.

Table 23-24: Geometric Mean of BWRX-300:UK ABWR by Radionuclide Class for Design Basis Source Term

PST	RN Class 1	RN Class 2	RN Class 3	RN Class 4	RN Class 5	RN Class 6
Water	N/A	5.19	22.56	1.10	0.56 [1] 0.28 [2]	0.72
Steam	5.15	5.17	22.56	1.92	0.56 [1] 0.28 [2]	0.72

Notes:

- BWRX-300:UK ABWR Radionuclide Class 5 tritium DB source term ratio without any adjustment to UK ABWR to account for zero-release strategy.
- BWRX-300:UK ABWR Radionuclide Class 5 tritium DB source term ratio with 2.0x adjustment to UK ABWR to account for zero-release strategy.
- Radionuclide Class 6 only contains FPs for DB comparison as AP concentrations are not substantially affected by fuel failure.
- The UK ABWR ‘DB’ PST has been adjusted using the methodology detailed in ANSI/ANS 18.1-2020 to allow for comparison.
- Only radionuclides that are included in both the UK ABWR PST and in ANSI/ANS 18.1-2020 have been included in the comparison. 12 nuclides in Radionuclide Class 1, 5 nuclides in Radionuclide Class 2, 4 nuclides in Radionuclide Class 3, 1 nuclide in Radionuclide Class 4, 1 nuclide in Radionuclide Class 5 and 15 nuclides in Radionuclide Class 6.

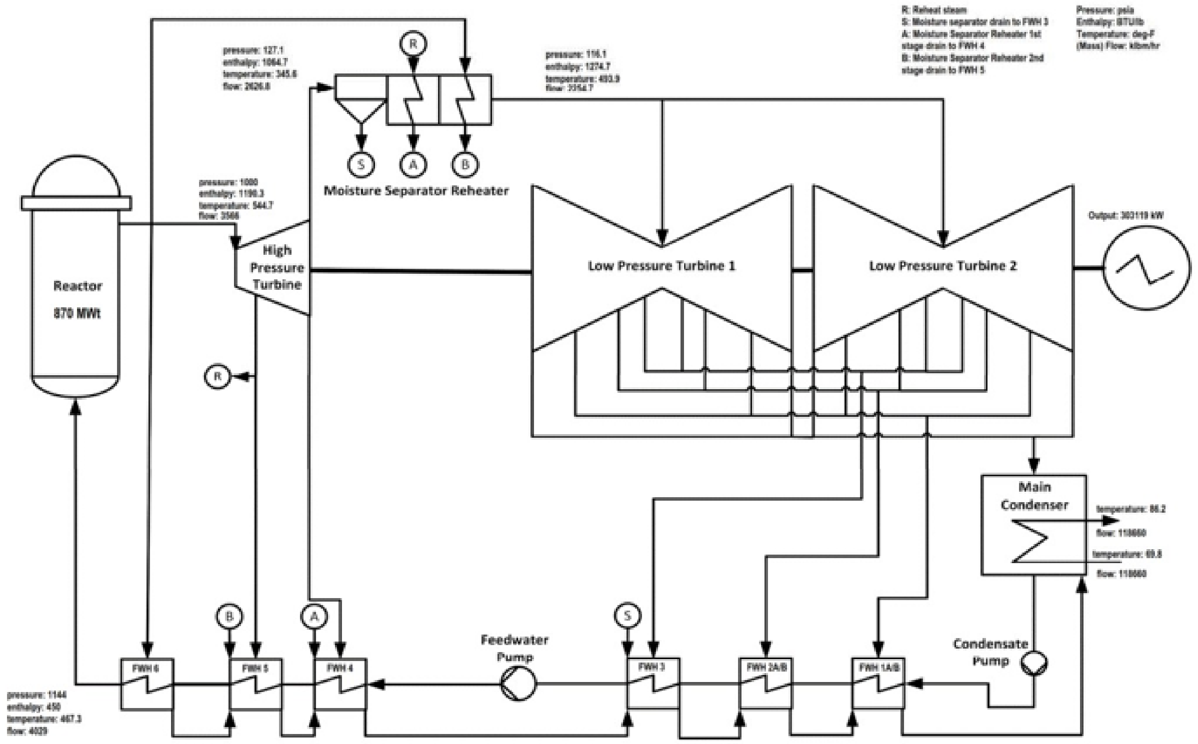

Figure 23-1: Flow Diagram Showing Main Systems of the Reactor Cooling Circuit

Legend:

DBR-0078661, BWRX-300 CRN Turbine Heat Balance in Support of TVA PSAR (Reference-23-21)

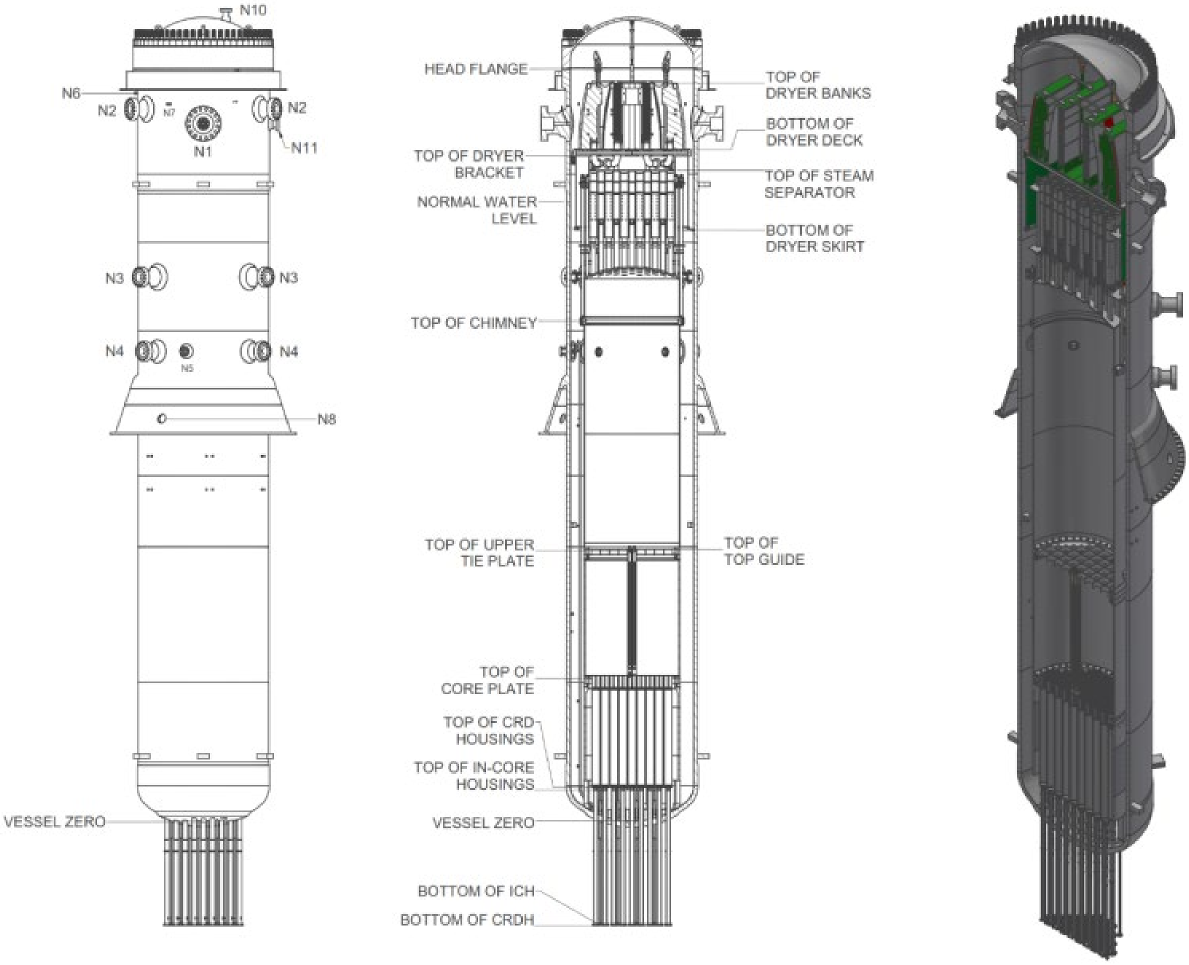

Figure 23-2: BWRX-300 Reactor Pressure Vessel and Internals

Legend:

006N7828, “BWRX-300 Nuclear Boiler System, System Design Description,” (Reference 23-22)

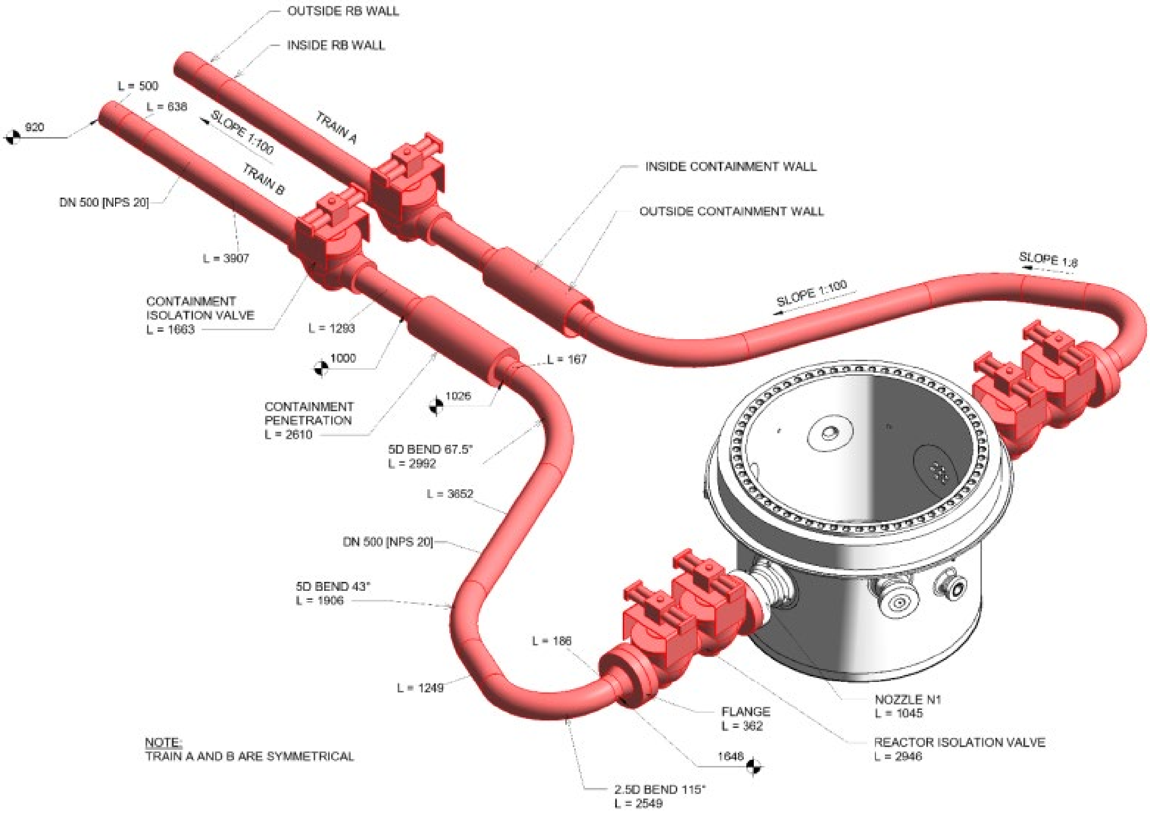

Figure 23-3: Main Steam Piping

Legend:

006N7828, “BWRX-300 Nuclear Boiler System, System Design Description,” (Reference 23-22)

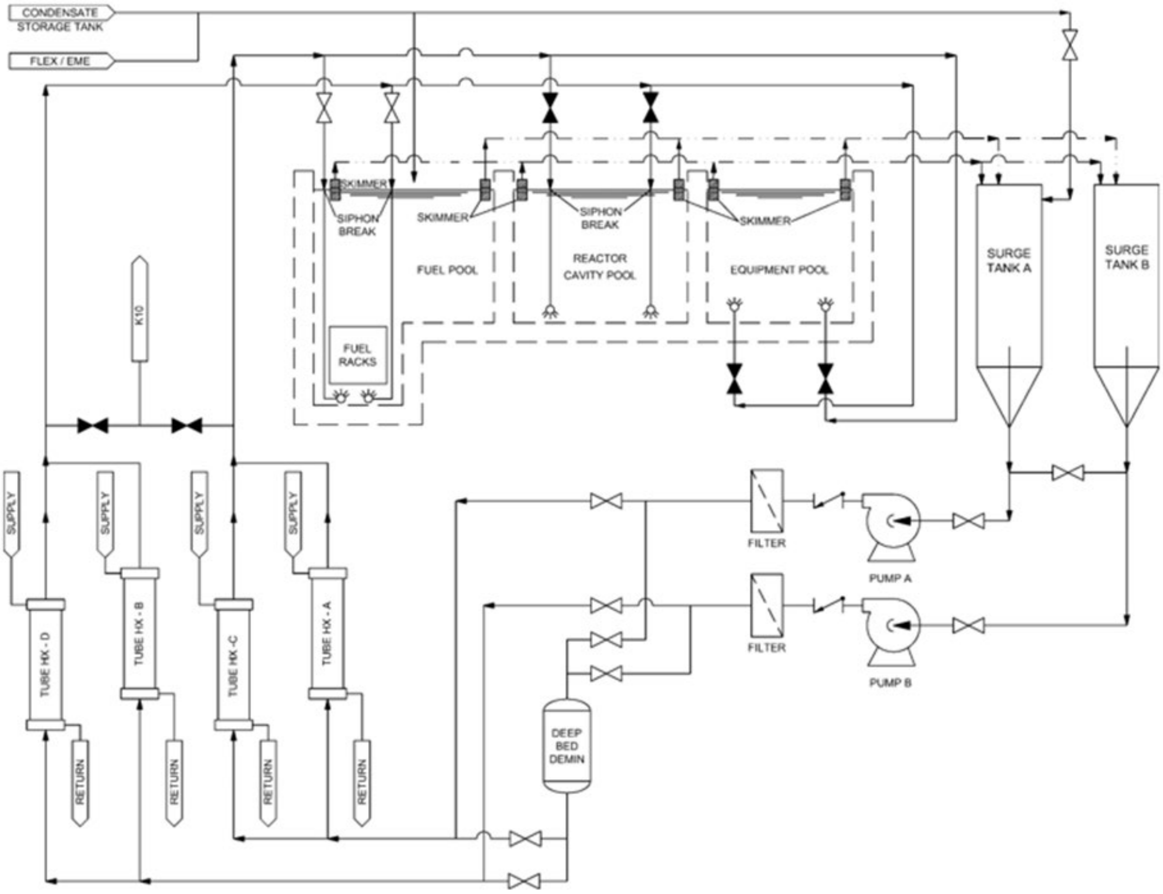

Figure 23-4: Fuel Pool System

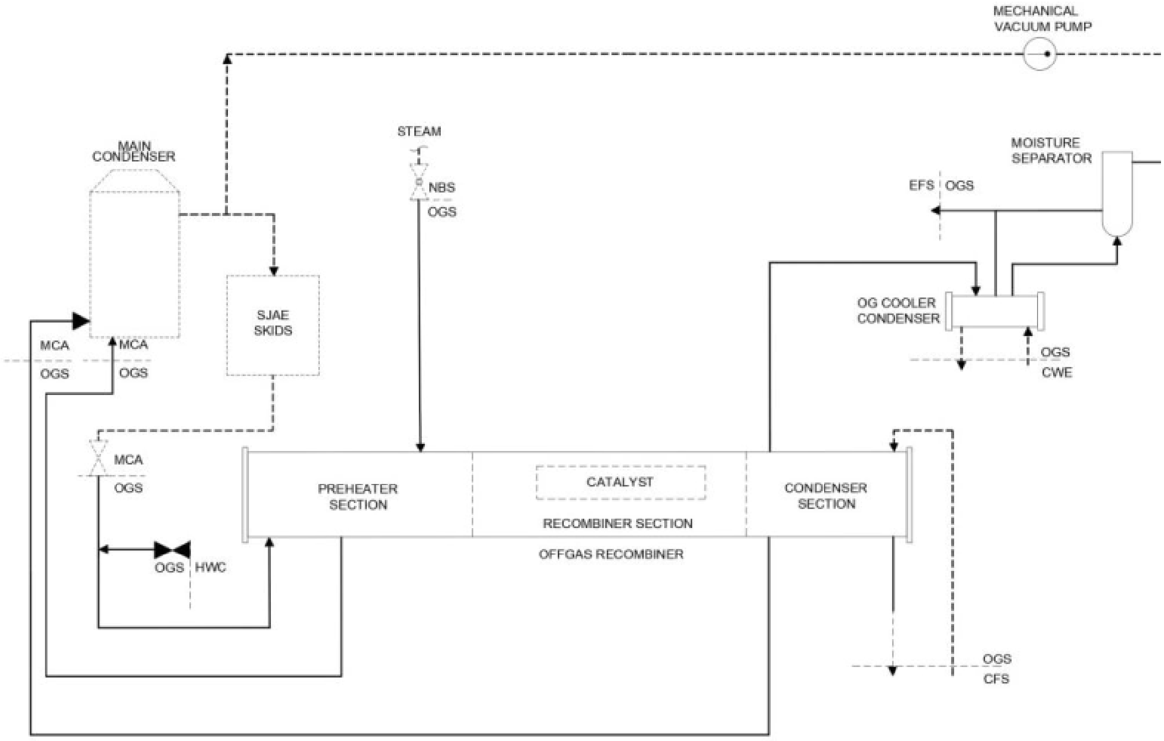

Figure 23-5: Part of the Offgas System Showing the Three Stage Offgas Recombiner

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APPENDICES

Additional Appendices are provided covering the following topics:

Appendix A: Claims, Arguments and Evidence.

Appendix B: Forward Action Plan.

Appendix C1: Historical Evolution of BWR Chemistry.

Appendix C2: Chloride Ingress Management.

Appendix C3: Commissioning Chemistry.

Appendix C4: Accident Chemistry.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

Claims, Arguments, Evidence

The SAPs 2014 (Reference 23-15) identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The Claims, Arguments and Evidence (CAE) approach can be explained as follows:

- Claims (assertions) are statements that indicate why a facility is safe
- Arguments (reasoning) explain the approaches to satisfying the claims
- Evidence (facts) supports and forms the basis (justification) of the arguments

The GDA CAE structure is defined within NEDC-34140P, BWRX-300 UK GDA Safety Case Development Strategy," (SCDS) (Reference 23-51) and is a logical breakdown of an overall claim that:

> *"The BWRX-300 is capable of being constructed, operated and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK".*

This overall claim is broken down into Level 1 claims relating to environment, safety, security, and safeguards, which are then broken down again into Level 2 area related sub-claims and then finally into Level 3 (chapter level) sub-claims.

The Level 3 sub-claims that this chapter demonstrates compliance against are identified within the SCDS (Reference 23-51) and are as follows:

2.1.5. Ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance and testing will be specified to maintain systems/ structures fit-for-purpose through-life

2.4.1 Relevant Good Practice has been taken into account across all disciplines

2.4.2 Operational Experience and Learning from Experience (LfE) has been taken into account across all disciplines

In order to facilitate compliance, demonstration against the above Level 3 sub-claims, this PSR chapter has derived a suite of arguments that comprehensively explain how their applicable Level 3 sub-claims are met (see Table 23-A-1).

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments, as this is beyond the scope of GDA Step 2. However, where evidence sources are available, examples are provided.

Risk Reduction As Low As Reasonably Practicable

A fundamental objective to be demonstrated through the GDA, is that the proposed BWRX-300 could be commissioned, operated, and decommissioned in the UK on a site bounded by the generic site envelope while maintaining the safety and security of people and the environment. A key component of this demonstration is that risks to operators and members of the public have been reduced to ALARP. This holistic high-level ALARP justification of the BWRX-300 is presented in NEDC-34199P, "BWRX-300 UK GDA PSR Ch. 27: ALARP Evaluation," (Reference 23-52).

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a 2-Step GDA. It is considered that the most that can be realistically achieved is to provide a reasoned justification that the BWRX-300 Small Modular Reactor (SMR) design aspects will effectively contribute to the development of a future ALARP statement. In this respect, this chapter contributes to the overall future ALARP case by demonstrating that:

The chapter-specific arguments derived may be supported by existing and future planned evidence sources covering the following topics:

- RGP has demonstrably been followed
- OPEX has been taken into account within the design process
- All reasonably practicable options to reduce risk have been incorporated within the design

It supports its applicable level 3 sub-claims, defined within the SCDS (Reference 23-51).

The management and control of chemistry in the various systems of the BWRX-300 is important for the safe and reliable operation of the plant and to minimise its environmental effects. The chief aim is to achieve continued structural integrity of the structures, systems, and components, minimise fuel failures and limit the generation and transport of radioactive material (i.e., the source term) within the plant. By reducing the source term SFAIRP, doses to workers and the public will be minimised. In addition, radioactive waste accumulation and routine airborne and liquid effluent discharges will also be reduced.

The BWRX-300 chemistry regime builds on OPEX and RGP obtained from the historical evolution of BWR chemistry developed over many decades of operation. The chemistry regime is bespoke to the BWRX-300 design and is an integral part of maintaining plant condition. In particular, the integrity of the reactor cooling circuit SSCs and the fuel by chemical dosing and impurity control (both within specified limits) and material condition. For other SSCs, the chemistry regime will contribute to the maintenance of integrity by combinations of chemical dosing, impurity control and materials selection. In addition, the BWRX-300 chemistry regime will ensure that the source term radiological dose to the workers and public is minimised and that the source term reduces waste accumulation and routine discharges. The chemistry regime achieves these objectives by optimising material selection and through operating practices.

Table A-1: Reactor Chemistry Claims and Arguments

Level 23 Chapter Claim	Chapter 23 Argument	Subsections and/or Reports that Evidence the Arguments	
2.1 The functions of systems and structures have been derived and substantiated taking into account RGP and OPEX, and processes are in place to maintain these through-life.			
2.1.5. Ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to maintain systems/structures fit-for-purpose through-life.	a) The BWRX-300 chemistry regime will control all corrosion-related degradation mechanisms that are important for fuel and SSC integrity.	23.1.3	Chemistry Dosing
		23.1.4	Water Purification
		23.1.5	Chemistry Parameter Ranges and Limits
		23.1.6	Sampling and Measurement
		23.1.7	Materials Selection
		23.1.8	Chemistry Management and Control during Reactor Startup and Shutdown
		Appendix C2	Chloride Ingress Management
		Appendix C3	BWRX-300 Commissioning Chemistry
2.4 Safety risks have been reduced as low as reasonably practicable.			
2.4.1 RGP has been taken into account across all disciplines.	a) The BWRX-300 chemistry regime will contribute to the maintenance of integrity of the reactor cooling circuit SSCs by chemical dosing and impurity control (both within specified limits), and by material selection.	23.1.3	Chemistry Dosing
		23.1.4	Water Purification
		23.1.5	Chemistry Parameter Ranges and Limits
		23.1.6	Sampling and Measurement
		23.1.7	Materials Selection
		23.1.8	Chemistry Management and Control during Reactor Startup and Shutdown
		Appendix C2	Chloride Ingress Management
		Appendix C3	BWRX-300 Commissioning Chemistry
		23.1.3	Chemistry Dosing

Level 23 Chapter Claim	Chapter 23 Argument	Subsections and/or Reports that Evidence the Arguments	
	b) The BWRX-300 chemistry will contribute to the maintenance of integrity of the fuel by chemical dosing and impurity control (both within specified limits), and by material selection.	23.1.4	Water Purification
		23.1.5	Chemistry Parameter Ranges and Limits
		23.1.6	Sampling and Measurement
		23.1.7	Materials Selection
		23.1.8	Chemistry Management and Control during Reactor Startup and Shutdown
	c) The BWRX-300 chemistry regime will contribute to the maintenance of a range of other SSCs by combinations of chemical dosing, impurity control and materials selection.	23.2	Fuel Pool System
		23.3	Plant Cooling Water System
		23.4	Isolation Condenser Systems
		23.5	Passive Containment Cooling System
	d) The BWRX-300 chemistry regime will ensure that the source term radiological dose to the worker and public is ALARP by optimising materials selection, operating chemistry, and operating practices	23.8	Radioactive Source Term
		23.8.1	Reduction of Source Term SFAIRP
		23.8.4	Primary Source Term - Validation
	e) The BWRX-300 chemistry regime will ensure that the source term reduces waste accumulation and routine discharges by optimising materials selection, operating chemistry, and operating practices	23.1.3	Chemistry Dosing
		23.1.7	Materials Selection
		23.8.1	Reduction of Source Term SFAIRP
2.4.2 OPEX and LfE has been taken into account across all disciplines.	a) The BWRX-300 chemistry regime builds on the historical evolution of BWR chemistry developed over many decades of operation	Appendix C1	Historical Evolution of BWR Chemistry

APPENDIX B FORWARD ACTION PLANS

Table B-1: Reactor Chemistry Forward Actions

Reference	Findings	Forward Actions	Delivery Phase
PSR23-132	The approach to controlling the iron concentration in FW needs to be elaborated.	Due to the adoption of different design features and materials in the BWRX-300, the iron FW limits, and control philosophy previously proposed for UK ABWR will not be transferrable. BWRX-300 will not be recommending an iron FW concentration range of 0.1ppb – 1.0ppb but will aim to operate with an iron FW concentration as low as possible. Furthermore, since the UK ABWR limits were justified on ALARP grounds, additional arguments and evidence will be needed to support this new position.	During PCSR development
PSR23-133	A considerable expansion in the detail of source terms will be needed for later stages of regulator assessment.	Although GEH has presented detailed Realistic Model (RM) and DB PSTs for reactor water and reactor steam, and some limited PrST to aid shielding assessments, other EUSTs derived for: 1) waste accumulation 2) liquid/airborne discharges are overly conservative and need to be re-evaluated. In addition, there is no systematic derivation for PrST for all plant regions**. This will be addressed.** Neither is there a systematic derivation of a DST to account for buildup of radioactivity on pipes and equipment, for the same plant regions. **This will be addressed.** Currently, there is no account taken for transient spikes in CPs and FPs that are known to occur at shutdown and in corrosion product spikes at startup are affected. **This will be addressed.** Transient and steady state source terms **will be combined** to derive CA source terms. Finally, EUST **will be developed** for the following discipline areas: Radiation protection (shielding and dose) Radioactive waste Decommissioning Normal discharges Fault studies	During PCSR development

Reference	Findings	Forward Actions	Delivery Phase
		All EUST will be derived using different combinations of BE, DB, PST, PrST, DST, CA source terms.	
PSR23-133	Provide further information relating to commissioning chemistry.	GEH to develop a strategy of how it intends to undertake commissioning chemistry. This is likely to include: defining and providing on-site monitoring of water chemistry, foreign materials exclusion, surface cleanliness, materials compatibility, and flushing operations during Cold Hydrostatic Testing (CHT) and oxide film conditioning during Hot Functional Testing (HFT). It is recommended that the project obtains the Topic Report on Commissioning Chemistry submitted during UK ABWR GDA. This can form the basis of an equivalent BWRX-300 Commissioning Chemistry Document.	During PCSR development
PSR23-333	The need for inclusion of a Chloride Ingress Protection System (CIPS) for UK based, sea water adjacent BWRX-300 to be finalised	Although not to be included in the base design for BWRX-300, GEH will not foreclose the option to include a CIPS but will defer further consideration until the site licensing stage. **This will be addressed.**	Alongside assessment for specific site

APPENDIX C1 HISTORICAL EVOLUTION OF BWR CHEMISTRY

C1.1 Introduction

Over the past few decades, the water chemistry of BWRs has evolved from using pure, relatively oxidising water to now including chemical dosing systems as summarised in: BWR Source Term Management - Strategies and Results at General Electric-Designed BWRs (Reference 23-53). This includes:

- HWC to mitigate IGSCC (Reference 23-53)
- DZO addition to minimise shut down dose rates (Reference 23-53)
- Noble Metal Chemical Addition (NMCA) or OLNC to optimise the HWC regime to limit the amount of hydrogen added to the system and as a result to reduce MS operating dose rates (Reference 23-53)

Advances in BWR Water Chemistry (Reference 23-54) with time is shown in Figure C1-1. Along with the chemical dosing regime, impurities such as iron, copper, chlorides, and sulfates are controlled to levels considered RGP (Reference 23-54).

Reactor water comes into contact with a variety of materials that comprise the plant SSCs, including nickel-based alloys, stainless steel, low-alloy steel, zirconium-based alloys of fuel rods, carbon steel and cobalt-based alloys. Thus, the management of reactor water chemistry is important to:

1. Maintain the integrity of the fuel
2. Maintain material and structural integrity
3. Reduce the dose rate experienced by workers and the public
4. Minimise radioactive waste produced
5. Prevent and or minimise radioactivity release to the environment

Structural integrity was identified as a source of concern when the evolution of Stress Corrosion Cracking (SCC) in BWRs was observed in the 1960s. In the early days of BWR operation, instead of chemically dosing the reactor water, any IGSCC concerns were addressed by replacing the effected material with improved IGSCC resistant materials, performing repairs, or chemical cleaning during outages at some BWR plants. While these measures were efficient in minimising the radioactive dose, they were expensive and time demanding, resulting in a longer critical path time, see “Hydrogen Water Chemistry for BWRs: a Status Report on the EPRI Development Program,” (Reference 23-55).

The purpose of this section is to detail the historical evolution of water chemistry in BWRs as it developed over time. Section C1.2 covers NWC, and the key issues encountered during early BWR operation, and Sections C1.3–C1.5 provide detail on how the issues associated with NWC were addressed and managed with new chemistry regimes.

C1.2 Normal Water Chemistry

During the 1950s, the primary goal of BWR Water Chemistry control was to manage conductivity and chloride levels in order to avoid TGSCC of SSCs and to limit the growth of scale on fuel rods. Table C1-1, taken from: “Introduction to Boiling Water Reactor Chemistry, LCC7 Special Topic Report,” (Reference 23-56), summarises the specification relating to NWC.

Over time it became clear that operating BWRs under NWC conditions many plants suffered IGSSC issues. By 1974, small and large diameter pipes manufactured from annealed type 304 stainless steel developed IGSCC, resulting in expensive inspection, repair, and replacement of piping in a number of plants. Laboratory investigations confirmed the link between stainless steel IGSCC failure in specimens and exposure to simulated BWR oxidising conditions and contaminants in reactor water, such as chlorides, sulfates, and organics. Early mitigation attempts included reducing residual stress in existing pipes using induction heating or mechanical stress management techniques, as well as replacing old pipes with type 316 stainless steel (Reference 23-57), (Reference 23-58). Hydrogen injection technique was later developed as a method of reducing IGSCC, and the historical progression steps are explained in Section C1.3 (Reference 23-53).

Another major concern effecting plant operation due to NWC was that there was significant development of fuel cladding defects. It was proposed that insufficient water chemistry control, such as excessive suspended iron crud concentrations, was one of the causes. Fuel cladding problems, in turn, caused additional aspects of poor water chemistry, such as the leakage of active FPs into the coolant (Reference 23-58). The key events and timeline are shown in Figure C1-2.

C1.3 Hydrogen Injection and Noble Metal Addition

The most significant contaminants in NWC that contributed to IGSCC where oxidising species formed from the radiolytic decomposition of water and other contaminants, including chlorides, organics, carbon dioxide, silicon dioxide, sulfates, nitrates, fluorides, and borates. Anions including chlorides and sulfates were postulated to enter crack tips, lowering the ECP and causing crack propagation. Other contaminants influenced the stability of the protective passive film on austenitic stainless steel, “The Role of Water Chemistry for Environmentally Assisted Cracking in Low-Alloy Reactor Pressure Vessel and Piping Steels under Boiling Water Reactor Conditions,” (Reference 23-59), and Water Chemistry in Boiling Water Reactors,” (Reference 23-60).

C1.3.1 Normal Water Chemistry to Hydrogen Water Chemistry

Under Normal Water Chemistry (NWC), the reactor water is oxidising and give a large ECP value of between -0.05 to 0.2 V. However, it was observed that by injecting hydrogen into the reactor water the concentration of oxidising species that are formed from the radiolysis of water such as hydrogen peroxide and oxygen can be suppressed. A decrease in the concentration of these oxidising species will result in a more reducing environment that will, in turn, reduce the ECP value to below -0.23 V and prevent IGSCC, “A Review of Stress Corrosion Cracking of Austenitic Stainless Steels in PWR Primary Water,” (Reference 23-61). The key chemistry changes in going from NWC to HWC are summarised in Table C1-3.

However, at FW hydrogen concentrations of 1 to 2 ppm, which are required for IGSCC mitigation of reactor internals, MS line radiation levels were found to increase by up to five times the level without hydrogen injection. The primary source of increased steam radiation is N-16 (7.1 second half-life), which is present as volatile forms (16NH3 and 16N2) in the reducing chemistry environment created by HWC. This raised concerns since the piping N-16 dose rates would affect workers' dose rates during plant power operation, “Methods to Control N-16 Radiation Levels in Steam Phase BWRs,” (Reference 23-62).

C1.3.2 Low Hydrogen Water Chemistry and Noble Metal Chemical Addition

NMCA involves the deposition of small amounts of catalytic material, such as platinum and rhodium, on the wetted surfaces in contact with the reactor coolant to catalyse recombination reactions of hydrogen with oxidants at those surfaces. In 1996, a study was carried out at the Duane Arnold reactor plant to determine the amount of hydrogen gas required to obtain the specified ECP value with and without NMCA. Without NMCA, a hydrogen injection concentration of 1 ppm was employed, but with NMCA, 0.2 to 0.4ppm was required, with no influence on MS line radiation levels. Therefore, a major advantage of noble metal application is that there is little to no increase in MS line radiation from N-16 activity at these low hydrogen injection rates. Studies indicated that increasing the hydrogen injection concentration from 0.4 to 0.5 ppm (hydrogen injection concentration at full plant power) resulted in 2.5 times increase in N-16 dose rate. It is therefore considered RGP to function with hydrogen FW concentrations of 0.2 to 0.4 ppm while using the noble metal chemistry regime, “Noble Metal Chemical Addition from Development to Commercial Application,” (Reference 23-63).

C1.3.3 Low Hydrogen Water Chemistry and On-Line Noble Chemistry

Following NMCA, GE developed OLNCTM technology, which was used as an alternate means of delivering noble metal to surfaces. OLNC injects only platinum when the plant is operating at or near full power, and reapplication is recommended every 11 to 16 months. Similarly to NMCA, OLNC only requires low FW hydrogen concentrations to achieve low ECP, and operational dose benefits are realised as with NMCA. The major benefit to OLNC over NMCA is the ability to apply the noble metal without shutting the reactor down, other differences between NMCA and OLNC are highlighted in Table C1-4, “Noble Metal Chemical Addition for Mitigation of Stress Corrosion Cracking,” (Reference 23-64).

C1.4 Zinc Addition

A relatively recent advancement in water chemistry is the use of zinc addition. The main aim of zinc injection is to manage radiation field build-up by lowering the dose of soluble and deposited Co-60 and Co-58, “Key Issues in Plant Chemistry and Corrosion in BWRS,” (Reference 23-65). Hope Creek was the first BWR to use Natural Zinc Oxide (NZO) for zinc addition in 1987. During reactor shutdown, when the FW temperature reached 176°C, a meter pump was used to inject a slurry of NZO at a concentration of 10-15 µg/kg. During the outage/maintenance phase, the zinc concentration in reactor water was reduced down to 5-10 µg/kg, “BWR Dose Reduction Using Depleted Zinc Oxide, Including Effects of Noble Metal Chemical Addition,” (Reference 23-66). Three years after Hope Creek began using zinc injection, the form of zinc was changed due to unintended neutron activation of Zn-64 to Zn-65. Formation of Zn-65 resulted in considerably increased radioactive waste arisings and higher dose rates as identified in the BWR Radiation Assessment Level and Control (BRAC) programme. Zn-65 has a half-life of 244 days and emits a 0.58 MeV photon, “Depleted Zinc: Properties, Application, Production, Applied Radiation, and Isotopes,” (Reference 23-67). To avoid the generation of Zn-65, injection using DZO was adopted.

C1.4.1 Natural Zinc Oxide to Depleted Zinc Oxide

Zn-64 is the lightest of the five naturally occurring zinc isotopes and is two atomic mass units from the second lightest Zn-66, as shown in Table C1-5. This makes it possible to separate Zn-64 from its other isotopes forming DZO (Reference 23-65). The first use of DZO took place in September 1992 at the Kernkraftwek Leibstadt (KKL) BWR plant in Switzerland (Reference 23-60) and thereafter, NZO was phased out. The transition from NZO to DZO for the BWR fleet with time fleet is shown in Figure C1-2.

On modern plants, zinc injection is performed by taking a portion of heated FW flow and passing it over a bed of DZO pellets held in a tank. Slow dissolution of the pellets results in ionic zinc entering the FW, which is then returned to the RPV. The tank is sized to allow the loading of sufficient DZO pellets to last the entire reactor cycle. A typical DZO injection system is shown in Figure C1-3.

The Boiling Water Reactor Chemistry Performance Monitoring Report-2009 Edition (BWRVIP-190), published by the Electric Power Research Institute (EPRI), outlines good practice values. The current guidelines suggest injecting enough zinc to produce a Co-60:Zn-64 ratio.

C1.5 Control of Other Chemical Impurities

C1.5.1 Iron

FW iron in a BWR has an effect on the amount of CPs in the reactor coolant system and can lead to increased crud deposition on fuel. Past improvements to the condensate polishing system and the installation of filters upstream of the CF demineralisers have resulted in a drop in FW iron levels in recent years. It was originally considered RGP to keep FW iron levels between 0.1 and 1.0 ppm, “Boiling Water Reactor Chemistry Performance Monitoring Report,” (Reference 23-68). At that time recommendations for maintaining a lower bound for FW iron levels at 0.1ppb were made, since it was thought that with extremely low FW iron concentrations were responsible for increasing Co-60 coolant concentrations. However, with the transition to plants using more corrosion resistant materials and greatly reduced flow accelerated corrosion issues, there is an industry move to allow iron FW concentration below 0.1 ppb.

C1.5.2 Copper

Plants that used brass for condenser tubes reported an increase in copper levels in the reactor coolant, but as condenser tubes were replaced with titanium or stainless steel, the amount of copper reduced. The CF, CD, and CUW manage copper levels and maintain FW concentrations below 0.05 ppb. High concentrations of copper in reactor coolant have been linked to fuel corrosion at numerous plants (Reference 23-68). It is worthy of note that BWRX-300 does not have any copper in the primary system (Reference23-32).

C1.5.3 Chlorides & Sulfates

High concentrations of chloride and sulfates in the reactor coolant can create a corrosive environment, which leads to degradation mechanisms such as SCC, pitting and crevice corrosion of austenitic stainless steel. Chlorides and sulfates have always been a key focus for impurity control and the relevant SSCs such as the CF, CD and CUW help keep the concentration levels to below 100 ppb, “LWR Water Chemistry Guidelines,” (Reference 23-69).

C1.6 Conclusion

Water chemistry has evolved significantly in the past few decades, and it is now considered good practice for BWRs to implement chemical dosing systems such as HWC, NMCA/OLNC, and zinc addition while also maintaining control over impurities such as chlorides, sulfates, iron, copper, and organics.

Water chemistry in BWRs initially started out using pure demineralised water, with chloride contaminants and direct conductivity being controlled. This resulted in operational issues such as IGSCC, fuel cladding failures, increased radiation exposure for personnel, and the generation of significantly more radioactive waste. In the beginning, the intent was to replace the affected material with improved IGSCC resistant materials, perform repairs, or conduct chemical cleaning during outages.

However, in the 1980s, hydrogen injection was investigated to minimise the concentration of oxidising species formed from the radiolysis of water and lower the ECP value to below -0.23V to prevent IGSCC. However, this resulted in greater N-16 generation, and worker exposure. It was subsequently decided that NMCA would be utilised in conjunction with HWC to lower the quantity of hydrogen required to generate an ECP value appropriate for IGSCC prevention whilst also minimizing N-16 production. To replace NMCA, GE developed OLNCTM technology, which included injecting platinum into the plant over a 10 to 14-day period of operation. OLNC technology is currently widely utilised in industry to prevent IGSCC and is injected along with hydrogen.

The use of zinc to control the build-up of the Co-60/Co-58 radiation field is a recent development in the field of BWR Water Chemistry. This technology began by injecting NZO but was later discontinued due to the unintentional activation of Zn-64 to Zn-65. DZO containing significantly less Zn-64 was made by a centrifugation technique and has since displaced NZO for zinc injection.

BWRX-300 plans to use HWC, OLNC, and zinc addition, as well as the CFD system to manage iron, chloride, sulfate, and copper impurities. Many light water reactors have used these chemistry technologies, and operating experience data over the years indicates that these regimes help to manage fuel integrity, plant component integrity, reduce dose rate to workers, and help to minimise radioactive waste formation. As a result, it is considered RGP worldwide to apply these systems, and GE-Hitachi plans to operate the water chemistry of the BWRX-300 under this chemistry regime.

Table C1-1: NWC BWR Technical Specifications and Fuel Warranty Limits

Parameter	Typical Technical Specification	Typical Fuel Vendor Warranty Limits
	Feed Water (FW)	
Conductivity µS/cm	-	< 0.065
Total Fe µg/kg	-	< 15
Oxygen µg/kg	-	20 to 200
Copper µg/kg	-	< 0.5
Zinc µg/kg	-	< 0.6
	Reactor Water (RW)	
Sulfate µg/kg	-	< 100
Chloride µg/kg	<500	< 100
Conductivity µS/cm	<1.0	< 1.0

Table C1-2: Summary of Key Events in the Evolution of Water Chemistry

Year of Detection	Event	Water Chemistry and Radiation Control
1950-1980	IGSCC of stainless steel fuel cladding, alloy 182, low carbon 304 and 316. Accelerating Occurrence of IGSCC of BWR Internals observed in late 1970s. Radioactive exposure increases to workers.	Impurity control. Oxygen injection.
1980-1985	Iron crud reduction. BWR plants with low shutdown radiation levels.	Utilising low cobalt containing materials. Improving operational procedures of condensate polishing.
After 1985s	Cobalt-60 increase at every latest operating cycle	Zinc injection. Advanced Ni/Fe ratio control.

Legend:

“BWR Water Chemistry Guidelines – 2000 Revision, EPRI, BWRVIP-79, TR-103515-R2,” (Reference 23-57)

“Improvements of Water Chemistry for Moderating Environmental Impacts of BWR Plants,” (Reference 23-58)

Table C1-3: Summary of Typical BWR Water Conditions Under NWC and HWC

Parameter	Normal Water Chemistry	Hydrogen Water Chemistry
Temperature (°C)	274 to 290	
Pressure (MPa)	7.2	
Flow rate (m/s)	1-10	
pH	5.65	
Conductivity (µS/cm)	0.08 – 0.1	
Cl^- & SO_4^{2-}	< 1 ppb	
Composition	O_2, H_2O_2, H_2	Excess H_2
O_2	100 – 500 ppb	< 5 ppb
H_2O_2	50 – 1000 ppb	< 10 ppb
H_2	5 – 40 ppb	50 to 300 ppb
ECP (V)	-0.05 to 0.2	-0.5 to -0

Legend:

(Reference 23-59)

Table C1-4: Comparison Between Earlier NMCA and OLNC

Parameters	NMCA	OLNC
Operating mode	Hot shutdown	Run power > 85%
Injection points	Residual heat removal and recirculation system	FW system
Application temperature	125°C - 140°C	282°C - 288°C
Noble metals	Platinum and rhodium	Platinum
Application period	48 hours	10-14 days
Application frequency	4 – 4.5 years	12 – 16 months

Legend:

(Reference 23-63)

(Reference 23-64)

Table C1-5: Isotopic Composition of Natural Zinc and Depleted Zinc

Zn Isotope	Natural Zinc	Depleted Zinc
Zn-64	48.7	0.9
Zn-66	27.9	33.0
Zn-67	4.1	9.0
Zn-68	18.6	54.8
Zn-70	0.6	1.4

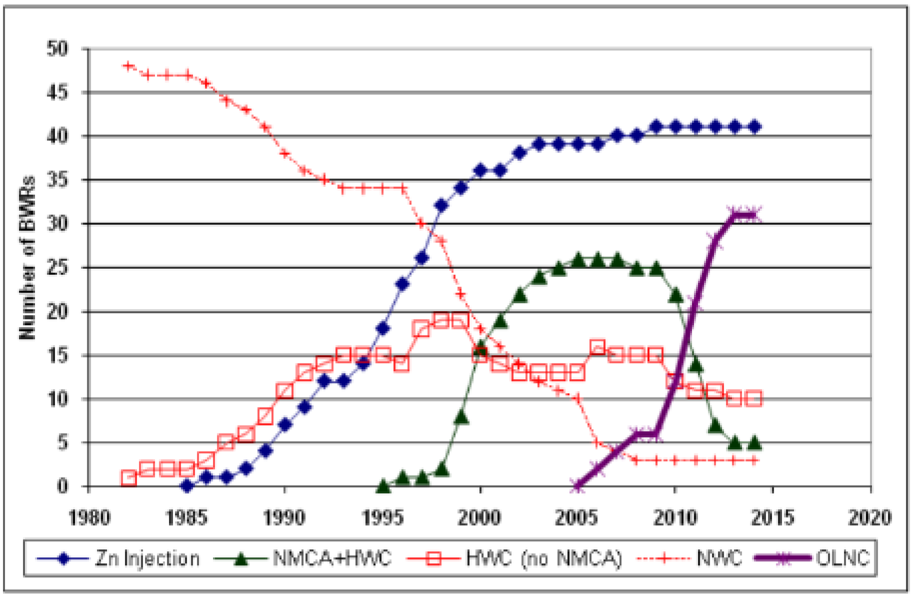

Figure C1-1: Chemical Dosing Systems

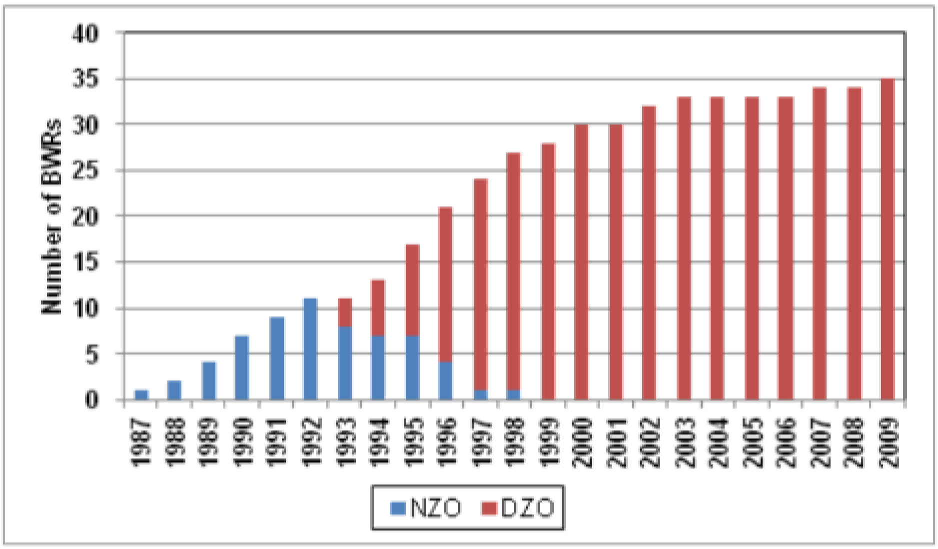

Figure C1-2: Plants Implementing NZO to DZO

Legend:

“Advances in BWR Water Chemistry (Reference 23-54)

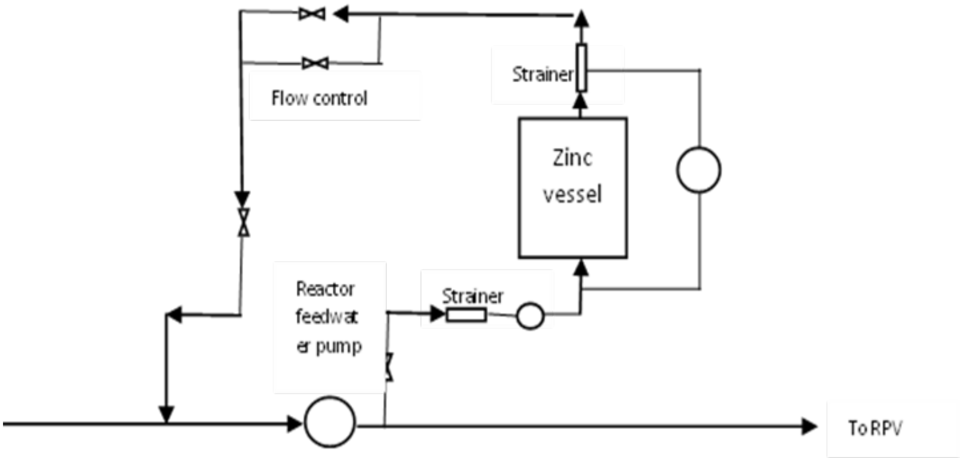

Figure C1-3: Depleted Zinc Oxide Injection System

APPENDIX C2 CHLORIDE INGRESS MANAGEMENT

During normal operation and when the condensate pump is in service chloride impurities will be managed via the CFD throughout Plant Modes 1 to 5. When the condensate pump is out of service and there are no leaks or faults that would lead to excessive chloride ingress, cleanup is performed as described in Section 23.1.4.

C2.1 Chloride Ingress Mitigation, Response and Recovery

In the event of a condenser tube leak that leads to ingress of water from an ultimate heat sink relying on seawater, chloride concentration in the FW will spike and FW will begin to leach chlorides over time depending on the size of the leak and resin capacity. Chloride ingress to the reactor FW poses corrosion risk to the system and in sufficient quantities for sufficient time at high temperatures significantly increases the potential for intergranular SCC of stainless steels and nickel-based components. Due to the risk posed by chloride ingress is it critical that any fault can be mitigated in design and, should they occur, that they are detected early, isolated and recovered to maintain continued safe operation of the reactor.

In order to mitigate the potential for chloride ingress via condenser tube leakage the condenser tubes will be made from highly corrosion resistant materials, either titanium or SEA-CURE® stainless steel. The condenser shells and hotwells are made of two cross-connected halves and their water boxes can be separately isolated, ensuring that minor leak repairs can be undertaken online with reduced power operation.

Another mitigation of the potential for chloride to migrate past the CFD following a minor tube leak is the selection of ion exchange resin in the CFD and inclusion of anion underlay resin in the system. Operationally, the potential for minor tube leaks to result in contaminated FW is mitigated by regular replacement of demineraliser beds well in advance of their saturation to ensure adequate capacity remains.

In order to ensure early detection and understanding of the scale of a condenser tube leak, conductivity is monitored in a number of places upstream of the CFD including the condenser hotwells and conductivity trays in the water boxes. In addition, conductivity is measured before and after the CFD to provide an understanding of the scale of any contamination downstream of the CFD. Action levels for measurements are clear and are based on RGP. The generic plant action levels (maximum values) for chloride is set at 100 ppb as indicated in Table 23-2. Plant-specific action levels will be established for condensate to prevent significant leakage into the reactor water.

Key chemistry parameter indications and alarms in the MCR allows for a range of responses in the event of chloride ingress that can be selected at various action levels:

- Isolation of a single water box into which the leak is occurring, and online repair in the event of minor leak detection
- Individual condenser isolation, and online repair in the event of minor leak detection
- Manual controlled shutdown triggered by alarms in the MCR to cool the reactor and mitigate damage should chloride ingress have the potential to overwhelm the CFD prior to major repair work

For manual or automatic shutdown, FW pumps will rapidly coast down and stop to prevent flow to the RPV. The ICS or SDC system can provide an alternative means of core cooling under these circumstances.

The BWRX-300 chloride management strategy is, therefore based around:

1. Design robustness to condenser tube failures
2. Early detection of ingress

3. Action levels with prescriptive manual and automatic responses
4. Eensuring any leak is recoverable by limiting any period where the reactor is at high temperature with high chloride content

C2.2 Chloride Ingress Protection System

The UK ABWR included in its design a dedicated Chloride Ingress Protection System (CIPS). CIPS was intended to act as an additional, automatic mitigation in the event of a major condenser leak, such as guillotine failure of one or more condenser tubes. The component was intended for use in the UK plant where the ultimate heat sink was seawater that could quickly overwhelm the CFD capacity for chloride cleanup. CIPS was designed to respond to conductivity sensors and meters that would be coupled to an automatic interlock system that could close the High-Pressure Condensate Pump (HPCP) downstream of the CF and CD. This system was intended to prevent contaminated condensate from being pushed forward to the primary containment vessel. When conductivity readings for both the CD inlet and CD outlet positions exceed the instrumentation logic set points, the CIPS interlock would be activated.

Beyond its inclusion in the UK ABWR design, no GEH designed BWR has ever operated with a dedicated CIPS and there is considerable OPEX for the mitigation and recovery of chloride ingress to provide confidence that such a system is not required for the BWRX-300. The combination of:

- Materials selection
- Early detection and continuous monitoring
- Capacity for isolation and repair of minor leaks online
- Manual and automatic scram in the case of action levels being reached, resulting in FW pumps rapidly coasting down and stop to prevent flow to the RPV,

all give confidence there is sufficient control to justify that the risk is managed SFAIRP. The inclusion of CIPS is not related to worker safety as relevant procedures have been put into place to keep workers safe (for example during water box interventions for repairs). Instead, GEH regard CIPS as an asset management system to prevent chloride attack of the plant's SSCs.

At this stage GEH will not foreclose the option to include a CIPS but will defer further consideration until the site licensing stage, see Forward Action Plan (Appendix B).

APPENDIX C3 COMMISSIONING CHEMISTRY

ONR has indicated the commissioning requirements in the SAPs (Reference 23-15), as follows:

- ECM.1: Before operating any facility or process that may effect safety it should be subject to commissioning tests defined in the safety case.

With particular regard to chemistry, Technical Assessment Guide (TAG) 088 (Reference 23-16) elaborates further:

- Commissioning can be a particularly important period for reactor chemistry, and this should be considered as part of the “adequate arrangements” required under Licence Condition 21. Control of chemistry during commissioning can have a significant affect on the subsequent safety of the plant. Appropriate commissioning tests should be carried out to ensure, for example, that design criteria, and where practicable, the safety functional requirements claimed within the safety case have been met (for example, dosing rates, purification performance, etc.,).

While it is recognised that the commissioning chemistry regime adopted for the BWRX-300 will be the responsibility of the licence holder, GEH, as vendor, intend to develop a chemistry commissioning strategy at a future stage of the UK licensing process. This commitment is captured in the Forward Action Plan (Appendix B). This strategy will set out a proposed chemistry commissioning plan for the license holder to follow to ensure that no activities undertaken during construction and commission will compromise downstream operations or plant safety.

At present, no options will be foreclosed. However, based on existing GEH experience and RGP, and what was proposed for UK ABWR, chemistry commissioning stages are expected to proceed as follows:

Before Fuel Loading

- Initial flushing operations with high quality demineralised water to clean all internal pipework and equipment
- Cold Hydrostatic Testing (CHT) of systems using high quality demineralised water

After Fuel Loading

- HFT, with progressive power increase in each consecutive commissioning cycle leading to full power
- During HTF, it is expected that water chemistry will be established with:
    - HWC at 5-10% power as early as practicable
    - Zinc injection to condition oxide films (to prevent future uptake of cobalt)
    - ONLC^{TM} to provide IGSSC protection prior to full power operation

The chemistry commissioning strategy is also likely to include consideration of:

- On-site monitoring of water chemistry
- Development of procedures to ensure foreign materials exclusion
- Maintaining surface cleanliness during reactor build, particularly for wetted reactor components
- Materials compatibility with process fluids in relevant reactor systems and developing procedures to provide materials compatibility testing pre-entry to site

APPENDIX C4 ACCIDENT CHEMISTRY

For the BWRX-300 there are currently no specific claims or implicit assumptions that regard chemistry minimising the effects of accidents.

The reactor relies on mechanical design of passive safety systems, such as the ICS and PCCS, avoiding the need for many of the traditional BWR engineered active safety systems that intervene in the event of Anticipated Operational Occurrences (AOO), Design Basis Accidents (DBA) and Beyond Design Basis Accidents (BDBAs). Furthermore, the inclusion of the ICS and BIS, a manual backup system used to inject enriched B-10 into the reactor, provides high resilience against occurrence of Severe Accidents (SA). Unlike other BWR designs, the BWRX-300 does not have a suppression pool. As such, in SAs involving fission product release in the dry containment volume there is no need for pH buffering of water to retard iodine release.

Credit, however, is taken for optimised chemistry, materials selection, and operational practices to enhance structural integrity and greatly reduce the likelihood of accidents occurring. The design focus is, therefore, on prevention of accidents and releases over limiting their severity.