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BWRX-300 UK Generic Design Assessment (GDA) Chapter 6 – Engineered Safety Features

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EXECUTIVE SUMMARY

The purpose of this Preliminary Safety Report (PSR) chapter is to describe the BWRX-300 Engineered Safety Features (ESFs) and its associated systems that are necessary to fulfil safety functions in the case of Design Basis Accidents, Design Extension Conditions, and some Anticipated Operational Occurrences. ESFs include:

- Isolation Condenser System
- Containment and associated systems (including Primary Containment System)
- Reactor and containment isolation valves
- Control Room habitability
- Boron Injection System

The chapter presents a level of detail commensurate with a two-step Generic Design Assessment (GDA) and is structured in line with the high-level contents of IAEA SSG-61.

System interfaces/dependencies are identified, and suitable cross references used to direct the reader to the relevant interfacing chapters of the safety justification.

Claims and arguments relevant to GDA Step 2 objectives and scope are summarised in Appendix A, along with an As Low As Reasonably Practicable (ALARP) position. Appendix B provides a Forward Action Plan (FAP), which includes future work commitments and recommendations for future work where 'gaps' to GDA expectations have been identified; however, there are no FAP items specific to this chapter.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
AC	Alternating Current
AHU	Air Handling Unit
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring Subsystem
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CAE	Claims, Argument & Evidence
C&I	Control and Instrumentation
CB	Control Building
CEPSS	Containment Equipment Piping Support Structure
CCS	Containment Cooling System
CFS	Condensate Feedwater Heating System
CMon	Containment Monitoring Subsystem
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CRD	Control Rod Drive
CRH	Control Room Habitability
CRE	Control Room Envelope
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
DBA	Design Basis Accident
DEC	Design Extension Condition
D-in-D	Defence in Depth
DL	Defence Line
DL2	Defence Line 2
DL3	Defence Line 3
DL4a	Defence Line 4a
DL4b	Defence Line 4b
DPS	Diverse Protection System
DPT	Differential Pressure Transmitter
DSA	Deterministic Safety Analysis

Acronym	Explanation
ECCS	Emergency Core Cooling System
EFCV	Excess Flow Check Valve
EFS	Equipment and Floor Drain System
EFU	Emergency Filter Unit
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
FAP	Forward Action Plan
FMCRD	Fine Motion Control Rod Drive
FSF	Fundamental Safety Function
FPS	Fire Protection System
FW	Feedwater
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HEPA	High Efficiency Particulate Air
HCU	Hydraulic Control Unit
HVS	Heating Ventilation and Cooling System
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	Isolation Condenser Cooling & Cleanup System
ICS	Isolation Condenser System
ILRT	Integrated Leak Rate Test
LfE	Learning from Experience
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
LWM	Liquid Waste Management System
MCR	Main Control Room
MSL	Main Steam Line
NBS	Nuclear Boiler System
NDT	Non-Destructive Testing
NPSH	Net Positive Suction Head
OLC	Operational Limits and Conditions
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PCS	Primary Containment System
PCCS	Passive Containment Cooling System

Acronym	Explanation
PIE	Postulated Initiating Event
PPS	Plant Pneumatics System
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
RB	Reactor Building
RBS	Reactor Building Structure
RCPB	Reactor Coolant Pressure Boundary
RGP	Relevant Good Practice
RIV	Reactor Isolation Valve
RO	Regulatory Observation
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SA	Severe Accident
SAP	Safety Assessment Principles
SCCV	Steel-Plate Composite Containment Vessel
SCDS	Safety Case Development Strategy
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDD	System Design Description
SSCs	Structures, Systems, and Components
TAF	Top of Active Fuel
TGFU	Toxic Gas Filtration Unit
TRACG	Transient Reactor Analysis Code General Electric
UK	United Kingdom
UPR	Ultimate Pressure Regulation
UPS	Uninterruptible Power Supply
USNRC	U.S. Nuclear Regulatory Commission
WGC	Water, Gas, and Chemical Pads

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Revision #	Section Modified	Revision Summary
A	All	Initial Issuance

6 ENGINEERED SAFETY FEATURES

The BWRX-300 incorporates ESFs, which mitigate the consequences of Anticipated Operational Occurrences (AOOs) or postulated Design Basis Accidents (DBAs) without any core damage. Information on the ESF materials is outlined in Section 6.1. Next, information on the Isolation Condenser System (ICS), which functions as the BWRX-300 Emergency Core Cooling System (ECCS), is captured in Section 6.2.1. The information on fission product containment and associated systems are presented in Section 6.5. Finally, the Control Room Habitability (CRH) function is outlined in Section 6.6.

ESF design features include passive systems that do not require dependence on external sources of power or operator actions fulfilling the Fundamental Safety Function (FSF). The BWRX-300 design does not require immediate operator responses for safety, as reactor shutdown with passive vessel and containment heat removal are automatically initiated. ESFs mitigate the consequences of accidents that may cause major fuel damage and cannot be safely shut down by the Safety Class 1 (SC1) function alone. Features include the prevention of radioactive material release and plant personnel protection from radiation exposure. The engineered safety features are:

- Isolation Condenser System, covered in Section 6.2
- Containment and associated systems including the reactor and containment isolation valves, covered in Section 6.5
- Control Room Habitability, covered in Section 6.6
- Boron Injection System, covered in Section 6.8

The BWRX-300 design is inherently safe, precluding the necessity for non-condensable gas control in the containment atmosphere following postulated DBAs. Design Extension Conditions (DECs) or Beyond Design Basis Accidents (BDBAs) are described and analyzed in PSR NEDC-34184P, "BWRX-300 UK GDA Ch. 15.6: Safety Analysis – Probabilistic Safety Assessment," (Reference 6-1). An overview of ESF materials selected in the BWRX-300 design is also included below.

The chapter presents a level of detail commensurate with a two-step GDA and is structured in line with the high-level contents of SSG-61.

System interfaces/dependencies are identified, and suitable cross references used to direct the reader to the relevant interfacing chapters of the safety justification.

6.1 Engineered Safety Feature Materials

Materials used in the ESF components ensure that material interactions do not impair ESF operation. The selection of materials used in ESF functions is based on existing, historical Operating Experience (OPEX) for Boiling Water Reactors (BWRs). Quality standards are used for the design, fabrication, erection and testing of ESF components using applicable codes and standards for the safety class identified. Materials are selected to withstand the environmental conditions encountered during normal operation and postulated accidents, including Loss-of Coolant Accidents (LOCAs). Material compatibility with core coolant water and the effects of radiolytic decomposition products are also evaluated.

The design, fabrication, erection and testing of the Reactor Coolant Pressure Boundary (RCPB) ensures a low probability of abnormal leakage or rapidly propagating failure or gross rupture.

Coatings used on exterior surfaces within the Primary Containment System (PCS) are suitable for expected environmental conditions.

6.1.1 Metallic Materials

Materials selected in the design of the BWRX-300 Nuclear Boiler System (NBS) builds on a history of BWR material OPEX. RCPB materials are provided in PSR NEDC-34167P, "BWRX-300 UK GDA Ch. 5: Reactor Coolant System and Associated Systems," (Reference 6-2), Section 5.2. Refer to PSR Ch. 5, Table 5-1, Typical Materials for Reactor Pressure Vessel Components, for the list of principal pressure-retaining materials and appropriate material specifications for the RCPB components. Refer to PSR Ch. 5, Table 5-3, Typical Reactor Internal Components and Materials for the principal materials and associated specifications for the reactor internal components.

Details on material selection for ESF Structures, Systems, and Components (SSCs), including details on metallic and organic materials used in the ESF systems, are compatible for the intended use.

All materials of construction used in ESF systems are resistant to corrosion, both in the medium contained and the external environment. General corrosion of all materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steel.

Demineralized water is employed in the Feedwater (FW) system. Refer to PSR NEDC-34173P, "BWRX-300 UK GDA Ch. 10: Steam and Power Conversion Systems," (Reference 6-3), Section 10.3 for a description of the water quality requirements. Based on OPEX, chloride leaching from concrete and other substances is not significant in BWRs.

No detrimental effects occur to any ESF materials from containment environment that affects the FW system. The materials are compatible with the post-LOCA environment.

6.1.2 Organic Materials

ESF equipment materials are selected considering the effects of radiolytic and pyrolytic decomposition and attendant effects on safe operation of the system. Refer to PSR NEDC-34175P, "BWRX-300 UK GDA Ch. 12: Radiation Protection," (Reference 6-4), Section 12.3 for ALARP design considerations.

Other organic materials in the containment are qualified to containment environmental conditions.

Evaluation of Materials for Engineered Safety Features

Materials are specified to withstand expected radiation doses during the plant lifetime. Non-metallic components such as gaskets and packing may have to changed out periodically as required by their respective qualified life.

Containment post-accident environment consists of hot water, nitrogen, and steam, no significant chemical material degradation is expected, due to strict inspection and testing. Solid debris from organic materials are unlikely.

6.2 Emergency Core Cooling Systems and Residual Heat Removal Systems

6.2.1 Isolation Condenser System (BWRX-300 Emergency Core Cooling System)

The System Design Description (SDD) for the ICS is 006N7492, "BWRX-300 Isolation Condenser System (E52) SDD," (Reference 6-5).

The BWRX-300 safety design philosophy for mitigating LOCAs credits conservative safety margins (e.g., larger water inventory), eliminates system challenges, and reduces the number and size of Reactor Pressure Vessel (RPV) nozzles relative to predecessor designs. In addition, all fluid system nozzles are located no less than four meters above the Top of Active Fuel (TAF). The large RPV volume, along with the tall chimney region, provides a substantial water reservoir above the core.

A Reactor Isolation Valve (RIV) assembly configuration is shown on Figure 6-1. These design features preserve reactor coolant inventory ensuring that adequate core cooling is maintained by isolation condensers removing the decay heat, following a LOCA.

Further details with respect to the Reactor Isolation Valves are provided within:

- The System Design Description (SDD) for the Primary Containment System (PCS), 006N7823, "BWRX-300 Primary Containment System (T10) SDD," (Reference 6-6)
- The System Design Description (SDD) for the Nuclear Boiler System (NBS), 006N7828, "BWRX-300 Nuclear Boiler System (B21) SDD," (Reference 6-7)

The large RPV volume also reduces the rate that reactor pressurization occurs if the reactor is isolated from its normal heat sink. If isolation occurs due to a design basis event, the hydraulic scram function initiates reactor shut down and the ICS, functioning as the ECCS, removes reactor heat. The slower pressurization rate, hydraulic scram function, and ICS eliminate the need for relief and safety valves for pipe breaks and RPV isolation events.

The ICS consists of three independent trains, each containing a heat exchanger or Isolation Condenser (IC) that is submerged in a dedicated pool of water that provides the ultimate heat sink for protecting the reactor core where the main condenser is not available or the RPV is isolated."

The single smallest inner pool volume, combined with the outer pool volume, provides reactor decay heat removal for a minimum of three days. The inventory of the two smallest inner pool volumes, combined with the outer pool volume, supports reactor decay heat removal for a minimum of seven days. Each inner and outer ICS pool is connected through two in-series SC1, one-way flow devices that prevent backflow from the inner pools to the outer pools.

The Isolation Condenser Cooling & Cleanup System (ICC) (NEDC-34171P, "BWRX-300 UK GDA PSR Ch. 9A: Auxiliary Systems," (Reference 6-8), in Section 9A.2.6) provides demineralized water makeup from the Water, Gas, and Chemical Pads (WGC) system which compensates for the relatively minor evaporative water inventory losses occurring under normal reactor operating conditions.

ICS removes decay heat generated in the core and does not require coolant injection into the RPV to mitigate pipe breaks and transients. Refer to NEDC-34183P, "BWRX-300 UK GDA PSR Ch. 15.5: Safety Analysis - Deterministic Safety Analyses," (Reference 6-9) in Section 15.5.4 for small steam and liquid pipe breaks inside containment.

6.2.1.1 System and Equipment Functions

The IC units located in the Reactor Building (RB) are submerged in an IC pool as shown in Figure 6-2 and Figure 6-3. The ICs condense steam on the tube side and heat is transferred to the IC pool water which boils, and steam is vented to the atmosphere. The ICS pool arrangement provides the ultimate heat sink for protecting the reactor core for any AOO or

DBA event where the main condenser is not available and the RPV is isolated. The IC is placed at an elevation above the steam source (RPV) so that this process is driven passively by gravitational force. When the steam is condensed, the condensate is returned to the RPV chimney via a condensate return pipe. The steam side connection between the RPV and the IC is normally open, and the condensate return line is normally closed. This allows the IC and condensate return piping to fill with condensate that is maintained at a subcooled temperature by the ICS pool water during normal reactor operation. This is 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 inventory while quenching steam and lowering pressure at the reactor core exit. 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.

Isolation Condenser System Pools

The ICS pools consist of three main or inner pools and three expansion or outer pools. The inner pools are train specific, in that each only supplies cooling water inventory to the corresponding IC located within the pool compartment. Each IC pool is separated by a reinforced structural wall so that the ICs are not vulnerable to common cause failure. The ICS pools bidirectionally communicate with each other above their respective weir elevations through the outer pools. The weir level maintains pool separation. If a failure occurs in an outer pool, the inner pool level is maintained at no lower than the weir height, thus maintaining the inner pool volume supporting the system SC1 function.

The outer pools are cojoined above and below the weir elevation sharing cooling water with all inner pools and ICs. Each outer pool shares nomenclature with the adjacent inner pool even though the outer pools provide cooling water to all inner pools. The outer pools:

- Physically protect the inner pools from outside effects.
- Provide water from all pools (inner and outer) to the ICC.
- Provide water inventory for a minimum of seven days for ICS decay heat removal, even when assuming one ICS train is unavailable.

Each outer ICS pool is connected to a cojoining inner ICS pool through doubly redundant, unidirectional, self-actuating flow devices that only permit flow from the outer pool to the inner pool, preventing inner pool inventory losses to a breached outer pool. The outer-to-inner pool one-way cross-connections are located within the pool compartments at an elevation that is below the IC tube vertical half-way point. This half-way point elevation defines the usable pools volume for boiloff heat removal capacity.

The simplified system diagram shown in Figure 6-4 illustrates the ICS pools arrangement. Note that “Inner Pool C” is the smallest of the three pools, as outlined in Figure 6-4. The minimum effective pool volume is down to the vertical half-way point of the IC tubes at an integrated reactor decay heat.

The ICS pools atmospheric vent piping originates from the ceiling of the ICS expansion pools and is routed vertically along the reactor building outside wall (refuel floor area) where it exits to the outside environment at an elevation below the polar bridge crane.

Steam Supply, Condensate Return and Standby Purge Piping

The steam supply, condensate return, and standby purge piping is designed to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), "Section III, Rules for Construction of Nuclear Facility Components, Subsection NB, Class 1 Components," (Reference 6-10). The steam supply and condensate return piping extends from the outboard RIVs to the respective connection points with the associated IC subcomponents. The standby purge piping extends from the connection point with IC standby purge line subcomponent to the outlet of the outboard standby purge isolation valve.

The condensate return piping is routed from the connection point with the IC vertically or with downward slope in its entirety to the condensate return valves and loop seal. From the condensate return valves and loop seal, it is routed vertically connecting with the RIVs where any subcooled condensate forming at the steam to loop seal interface gravity drains to the RPV. The condensate return piping stored water volume is a critical parameter credited in the safety analyses, resulting in design constraints placed on the inside diameter piping and total length. The dynamic head losses or pipe friction losses of the piping are important to the safety analyses and are modeled in Transient Reactor Analysis Code General Electric (TRACG) analysis that forms the safety basis for the ICS design of NEDC-33987P, "BWRX-300 Darlington New Nuclear Project (DNNP) TRACG Application," (Reference 6-11).

Isolation Condenser Description

The IC is designed to ASME BPVC Section III, Subsection NB (Class 1) (Reference 6-10).

Each ICS train has heat removal capacity of 33.75 MWt (3.8% of reactor thermal power). The known heat removal capacity of the ICs is established by full-scale prototype testing The ICS core decay heat removal capability at one minute is outlined in PSR Ch. 15, Section 15.5. IC heat removal capability increases when steam supply pressure increases, and it decreases when steam supply pressure decreases. The ICS trains have no common valves, piping, or RPV nozzles.

Condensate Return Valves and Loop Seal

Each ICS train has two 100 percent flow capacity condensate return valves installed in parallel. The two condensate return valves are remotely actuated with each designed to fail open upon a loss of control signal, control power, or pneumatic supply, as applicable to the valve actuator design. Both valves fail to the fully open position by stored spring energy. The valves are both located in horizontal runs of piping that are the lowest elevation of the condensate return piping, and the connection to the RIVs. This geometry creates a loop seal (discussed below) ensuring that the valves always have subcooled water on either side of their sealing surfaces, while the ICS is in standby, and while the reactor is in operation.

To place an ICS train in service, at least one of the two condensate return valves must open. The two valves installed in parallel ensure that no single active failure can cause the loss of any single train. The parallel valves and actuators are diversely designed, but both valves fail to the fully open position on a loss of pneumatics, control power, or control signal. Once the valves fully open, they stay fully opened until they are reclosed intentionally by the operator.

The water loop seal achieves two primary functions. First the loop seal performs a function under long-term operating conditions where the condensate return line liquid is low density and potentially in a two-phase state instead of a solid subcooled liquid column. The loop seal ensures a pocket of subcooled liquid exists in the condensate return line that prevents steam bypass conditions that would short circuit the system and lower heat removal performance. The vertical distance from the bottom of the loop seal to the RPV nozzle prevents steam bypass directly from the chimney region of the RPV to the isolation condenser by way of the condensate return line. Secondly, in the ICS standby condition, the loop seal ensures there is subcooled water on both sides of the condensate return valve seats that is nearly the same temperature. This eliminates valve body and seat distortion that could cause valve leakage.

Standby Gas Purge Piping and Isolation Valves

Whenever the ICS is in standby, and the reactor is critical, radiolytically-generated gases (hydrogen and oxygen) tend to accumulate in the upper point of the ICs where the steam to condensate interface exists within the steam distributor and associated piping subassembly. To prevent gas buildup eventually filling the distributor and the vertical steam supply line, a continuous gas purge is required. The purge line taps into the center region of the steam distributor and is routed back down into the Steel-Plate Composite Containment Vessel (SCCV) inside the penetration space between the vertical steam supply pipe and the outer guard pipe. The purge line is routed to the Main Steam Line (MSL) at a location between the RIV and the containment penetration.

The purge lines for ICS trains A and B connect to MSL A and the ICS train C purge line connects to MSL B. The pressure drop that occurs between the interior of the RPV and downstream of the RIV induces purge gas flow from the IC steam distributor to the MSL. Inside the SCCV, the gas purge line contains two automatic isolation valves installed in series that fail close on a loss of signal or power. The two in-series valve design provides redundancy in the event of a single active failure of one valve to close on demand. At least one of these valves close whenever the reactor is isolated, and the ICS is placed into service. The valve(s) close to prevent loss of inventory when the RPV is isolated.

The IC has an integrally mounted catalytic recombiner device that processes radiolytic gases when the system is in operation, eliminating the need to vent non-condensable gas. The catalytic recombiner does not alter the IC heat transfer characteristics.

6.2.1.2 Safety Design Bases

The BWRX-300 ICS is designed as a Defence Line (DL) 3 SC1 system that removes sensible and decay heat and provides RPV overpressure protection.

The ICS system meets the requirements of Safety Standards Series No. SSR-2/1, “Safety of Nuclear Power Plants: Design,” (Reference 6-13). The technical justification and U.S. Nuclear Regulatory Commission (USNRC) acceptance of the ICS functioning as the ECCS is provided in NEDC-33911P, “BWRX-300 Containment Performance,” (Reference 6-14). ICS DL functions during normal and off-normal reactor plant operating conditions are organized by plant safety functions. While the reactor is at normal power operations, the ICS is in standby with the pressure boundary intact and ready to initiate on either a manual or automatic initiation signal.

NEDC-33910P, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” (Reference 6-15) justifies the ICS RPV overpressure protection for the ASME Code Class 1 reactor pressure boundary. Refer to PSR Ch. 15, Section 15.5.4 for limiting DBA overpressure events.

In plant modes where the Shutdown Cooling System (SDC) is in-service, the ICS performs the SC1 Defence Line 3 (DL3) function of isolating the interface connection(s) to the SDC in the event a pressure boundary leak is detected. This function ensures that passive failure or misalignment in a lower safety class system cannot disable the higher safety class ICS. The function is most critical when the RPV head is still tensioned, making it an important plant mode-related function.

ICS electrical and mechanical equipment is qualified in accordance with the BWRX-300 Equipment Qualification Specification using the established environmental conditions at the ICS pools and associated equipment locations so that they remain functional during post-accident environmental conditions, in addition to the normal conditions. Refer to Table 6-1 for the ICS SSC Classification.

During normal plant operations, the ICS is in standby condition and performs the following safety functions:

- ICs and the condensate return lines are filled with water up to the steam distribution headers.
- Condensate return valves are closed, operable, and ready to open when ICS is initiated on a demand signal from interfacing Defence Line 2 (DL2), DL3, and Defence Line 4a (DL4a) control systems.
- Standby gas purge isolation valves are open venting gasses to the NBS main steam lines, operable, and ready to close when ICS is initiated on a demand signal from interfacing DL3 and DL4a control systems.
- ICS pools at or above their minimum level and at or below their maximum temperature.
- Check valves prevent backflow from each inner pool to the outer pool set are operable and preserve minimum inner pool inventory for DL2, DL3, and DL4a decay heat removal and RPV overpressure protection functions.
- When SDC is not operating, interface connections to the lower safety class SDC system are isolated and perform their DL3 and DL4a reactor coolant pressure boundary integrity function.

During shutdown plant operation, the ICS provides a suction path from the RPV chimney internal region to SDC for the DL2 decay heat removal function.

In response to AOO or DBA events, ICS safety functions are confinement of radioactive materials (through overpressure protection and RCPB integrity), fuel cooling, and long-term heat removal. These functions are performed responding to various DL initiating events:

- ICS provides overpressure protection by maintaining RPV pressure below design limits. The overpressure protection function is performed in conjunction with the hydraulic scram function for overpressure event scenarios that progress to DL3. ICS overpressure protection is active in response to:
  a) RPV operating pressure setpoints exceeded (DL2)
  b) RPV high pressure setpoints exceeded (DL3, DL4a)
- ICS performs reactor coolant inventory addition and decay heat removal functions in response to:
  a) RPV coolant level below setpoint (DL3 & DL4a)
  b) Containment high pressure setpoint exceeded (DL3 & DL4a)
  c) Line break indication (NBS main steam line, feedwater line, cleanup water, ICS) detected (DL3 & DL4a)
  d) Operator manually initiates ICS train(s) (DL4a).
  e) ICS initiation on Loss of Condenser Vacuum (DL2)
- When SDC is in operation, ICS maintains the integrity of the RCPB in response to a FW or SDC line break by automatically isolating the interface connection to the affected SDC train (DL3 and DL4a).

6.2.1.3 Description

The ICS removes sensible and core decay heat from the reactor passively without any loss of reactor coolant inventory when the main condenser is unavailable. Reactor heat is transferred from each IC heat exchanger to the IC pool water by condensation and natural circulation. No forced circulation equipment is required when the ICS is in-service following these events:

- Reactor isolation at power operating conditions
- Loss of all Alternating Current (AC)
- Failure to scram (BDBA as described in PSR Ch. 15, Section 15.6)
- LOCAs

Similar isolation condenser systems have been used in early BWRs (BWR/2 and BWR/3) as the emergency heat removal systems successfully for over 40 years. The genesis of the BWRX-300 IC is the Simplified Boiling Water Reactor IC design, which was tested extensively for performance and other operating characteristics. The same IC design is specified in the Economic Simplified Boiling Water Reactor (ESBWR). The ESBWR IC heat transfer capacity was increased or uprated by the addition of rows of tubes within the original array. This power-uprated ESBWR IC version is used in the BWRX-300. Other modifications in the BWRX-300 IC design include design pressure (higher for BWRX-300), return into the chimney instead of the downcomer (refer to Section 6.2.1.6), and hardware material selections made to enhance the design.

6.2.1.4 Materials

ICS electrical and mechanical equipment is qualified using environmental conditions where they are located. ICS components functioning under faulted conditions remain functional under the post-accident environmental conditions in addition to the normal conditions. The IC is a full-scaled prototype tested component. The seismic integrity of the ICS pools is discussed in NEDC-34172P, “BWRX-300 UK GDA PSR Ch. 9B: Civil Structures,” (Reference 6-16), Section 9B.2.3.

6.2.1.5 Interfaces with Other Equipment or Systems

Figure 6-5 identifies the ICS system interfaces and Table 6-2 summarizes the system interfaces. Each ICS train directly interfaces with RPV, and either SDC or the Boron Injection System (BIS) depending on the ICS train.

Interface Connections with Shutdown Cooling

ICS trains A and B each have two series isolation valves outside the containment penetrations in the interface lines to the SDC trains A and B (refer to PSR Ch. 9A, Section 9A.2.7 for more details of the Shutdown Cooling System) that provides a SDC suction path from the RPV chimney interior. The interface piping in the ICS is a connection located in the loop seal region between the condensate return valves and the RIVs. The interface piping is routed through a containment penetration and includes two in-series outboard remote actuated isolation valves. The actual ICS to SDC boundary occurs at the outlet of the most outboard isolation valve for each train.

The double isolation in-series valve design provides redundancy in the event of a single active failure to one valve. The isolation valve closest to the SCCV penetration has two safety class functions. One function is normal containment isolation where the valve works in tandem with its associated RIV. The other function is terminating inventory loss in the event of a SDC pressure boundary failure outside containment when the plant is shut down and relies on the SDC for decay heat removal and cooldown. For this function, the two outside isolation valves work in tandem, providing redundancy in the event of a single active failure. Even though the RIV and outside containment isolation valve can provide the same redundant protection against a SDC breech, the ICS train associated with that interface would not be functional for decay heat removal. The availability of a functional ICS train is especially critical while the RPV head is still installed or prior to flood up.

Refer to Section 6.5.6.1 for containment isolation functions of the SDC interface valves.

Interface Connection with Boron Injection System

ICS train C interfaces with the BIS in a similar fashion as the SDC. The system boundary isolation valves are part of the BIS, and the system interface occurs inside the containment building (refer to Section 6.8). The BIS system has one remote isolation valve outside containment and one check valve inside containment. The BIS system interfaces with the check valve inboard side.

6.2.1.6 System and Equipment Operation

Upon receipt of a RIV actuation, at least one condensate return valve in one ICS train opens and at least one standby gas purge isolation valve in each train closes. The subcooled condensate stored in the system during the standby state enters the RPV chimney interior providing additional inventory while quenching steam and lowering pressure at the reactor core exit. Simultaneously, steam from the RPV enters the IC where it condenses in the tubes and returns 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.

With ICS in operation (one, two or three trains as required), decay heat removal does not exceed the plant Operational Limits and Conditions (OLC) for RPV cooldown rate. One ICS condensate return valves in each train is used to throttle flow to maintain RPV pressure, producing the desired RPV cooldown rate. Eventually, one ICS train is sufficient to remove reactor decay heat. When the ICS is operating, the purge isolation valves close preventing the loss of RPV inventory. Radiolytic hydrogen and oxygen are recombined with passive catalytic recombiners installed integral to each of the ICs.

ICS control system functions are described below under Section 6.2.1.7.

Since the ICS is a closed system attached to a high purity reactor coolant system, it is not subject to flow blockages, particularly because the steam intake is from a high elevation in the RPV and the condensate discharge is to an elevation above the TAF.

An ICS train can only be isolated automatically if a detectable leak or break occurs and the particular train is not in operation.

6.2.1.7 Instrumentation and Control

The ICS provides emergency core cooling and is also the primary system preventing RPV overpressure. Diversity, redundancy, and backup power is supplied for the instrumentation and controls ensuring ICS operation when initiation occurs.

Steam Condensate Temperature Instrumentation

Each ICS train has temperature instrumentation installed on the steam supply line and condensate return lines. The temperature instrumentation installed on the steam supply lines is located as close to the containment penetration as practicable. Each IC will be provided with Safety Class 3 (SC3) temperature instrumentation installed on the upper headers, lower headers, and steam distributors that accurately measures the temperature of the bulk fluid within each header and reports these measurements to the control system platform.

Isolation Condenser System Pool Level and Temperature

Each ICS inner pool is equipped with narrow range level instrumentation at the weir elevation. Each ICS inner pool is equipped with level instrumentation that measures the wide range pool level while the ICS is in operation.

Each ICS inner pool is equipped with bulk pool temperature instrumentation.

RPV Isolation from ICS Following an ICS Line Break

The safety design basis assumes an ICS train failure occurs at the pressure boundary that necessitates quick closure of the RIVs for that train steam supply and condensate return lines to terminate a LOCA. In this event, the other two ICS trains remain intact and are not inadvertently isolated to mitigate the event (cool and depressurize the RPV). To ensure that inadvertent isolation of an ICS train does not occur at any time, two important features exist in the physical design and associated logic:

- The RIVs that interface with ICS are designed to fail as-is and are open under normal operating modes. This contrasts with all other RIVs, which have a fail-closed design.
- Redundant and diverse instrumentation and controls are provided.

The ICS RIVs closing function is described in NEDC-34169P, "BWRX-300 UK PSR GDA Ch. 7: Instrumentation and Control," (Reference 6-17), Section 7.3.

Each ICS train is equipped with two flow detection elements in the form of 90-degree bends that have impulse lines with three parallel Differential Pressure Transmitters (DPTs). One flow detection device is located in the steam supply line, and another located in the condensate return line. These DPTs detect flow within the system. If the ICS train is in standby and flow is detected, a line break is indicated and the control system for this line initiates RIV closure for this train only. ICS leak detection is not active during ICS operation, because the heat removal FSF is prioritised over LOCA mitigation and no automatic closure of RIV occurs. An ICS line leak occurring during ECCS FSF is a BDBA.

System Actuation, Opening the Condensate Return Valves and Closing the Gas Purge Isolation Valves

As indicated in Section 6.2.1.6, when an ICS initiation event occurs, two initiations occur that place the ICS into alignment for event mitigation:

- At least one condensate return valve in one ICS train opens.
- At least one standby gas purge isolation valve in each train closes.

The number of ICS trains placed into service depends on the event severity as a result of RPV pressurization rate. In the most severe event, condensate return valves in each train may open on a staggered basis. The parallel redundancy of the condensate return valves and the series redundancy of the standby gas purge isolation valves ensure that any single active failure does not disable any of the ICS trains.

The ICS operational logic places into service only the number of train(s) required to limit an RPV pressure increase and then holds pressure stable at a predetermined value, or lowers pressure at a prescribed rate, depending on the event circumstances. Both the modulating and full-open demand setpoints are staggered across all condensate return valves by train. The established setpoints modulate valves in all three trains initially limiting and then controlling RPV pressure with each train's setpoint incrementally higher than the lowest.

Should the RPV pressure fail to stabilize or continue increasing, setpoints are reached that trigger both condensate return valves in each train to fully open in an incrementally staggered sequence by train. Refer to PSR Ch. 7, Section 7.3.1.1.3, C10 SC1 Control & Instrumentation (C&I) Functions and Initiation Signals. The full open condensate return valve is a DL3 function (full open or close only valve) and a DL4a function controlled by the Diverse Protection System (DPS) control platform (both full open/close and modulating valves).

6.2.1.8 Monitoring, Inspection, Testing and Maintenance

The ICS is an SC1, Quality Group A and B system that is designed and constructed as ASME BPVC Class 1 and 2 (Reference 6-10 and 6-12). As such there are mandatory technical specification surveillance tests including in-service testing of components and in-service inspections of pressure boundary components such as piping and tube to header welds. The design and construction of the ICS integrates features that permit these tests and inspections to be performed. Note that in-service inspection of the tube to header welds necessitates opening of the headers and entry in order to perform volumetric testing from the inside diameter of the welds. General internal inspections of the ICs are to be performed as well.

OLC surveillance tests include in-service component tests and in-service pressure boundary component inspections (piping welds). The design and construction of the ICS integrates features that prevent thermal fatigue and permit tests and inspections.

Monitoring, inspection, testing, and maintenance are planned and performed to meet system reliability and availability targets. Refer to NEDC-34188P, “BWRX-300 UK PSR GDA Ch. 16: Operational Limits and Conditions,” (Reference 6-18), for a discussion on OLC.

Maintenance activities are expected to be minimal as the only active mechanical components are the two per train condensate return valves, the two per train standby gas purge isolation valves, the two per A and B train SDC interface containment isolation valves, and the Ultimate Pressure Regulation (UPR) component on each train which is on a prescribed periodic schedule.

The flow detection PDTs are also on a periodic maintenance and calibration schedule.

Periodic heat removal testing of each IC is performed on a rotating basis at an operating cycle interval.

6.2.1.9 Radiological Aspects

Large ICS breaks outside containment and small ICS line breaks are discussed in Chapter 15, Section 15.5.9.

The mass and energy release from the isolation condenser breaks outside containment are bounded by the isolation condenser pipe breaks inside containment that are described in PSR Ch. 15, Section 15.5.4. The isolation condenser pipe breaks do not use any DL3 functions that depend on containment parameters and the containment back pressure is not credited in any isolation condenser pipe break analyses.

As is the case for isolation condenser pipe breaks inside containment, core response is not a concern because the break is isolated rapidly. The consequences of large isolation condenser pipe breaks outside containment are evaluated for loads, pressures, and temperatures outside containment and radiological consequences resulting from normal operation coolant activity.

Since the small break analyses inside containment do not credit containment back pressure, the mass and energy release calculated for breaks inside containment are bounding for breaks outside containment. The ICS break analyzed for offsite dose consequences is a postulated break of an IC steam side component within the ICS pool compartment of the RB. The break is immediately detected and RIV isolation occurs in 10 seconds. Refer to PSR Ch. 15, Section 15.5.9 for ICS.

6.2.1.10 Performance and Safety Evaluation

In general, any one ICS train can mitigate AOOs. Two ICS trains are required for LOCA mitigation (analysis assumes one of the three ICS trains has a single failure). All three ICS trains are credited for beyond design basis events and the IC pool inventories can by replenished indefinitely. Functions of the ICS satisfy, the analysis assumptions for AOOs, and DBAs as described in PSR Ch. 15, Section 15.5. For BDBAs, refer to PSR Ch. 15, Section 15.6.

6.2.2 Residual Heat Removal Systems

Not within the scope of this chapter, see PSR Ch. 9A.2.

6.3 Emergency Reactivity Control System

The Control Rod Drive (CRD) system provides the primary means of reactivity control during normal, abnormal and accident conditions. The system design basis includes two diverse motive forces for the CRD insertion (scram) using high pressure water from the Hydraulic Control Units (HCUs), and control rod insertion using the Fine Motion Control Rod Drive (FMCRD) motor. Incorporated into the design are positioning and protective features that prevent inadvertent withdrawal, drop, and ejection of the control rod due to a component break or other malfunction. The CRD system is covered by NEDC-34166P, "BWRX-300 UK PSR GDA Ch. 4: Reactor (Fuel and Core)," (Reference 6-19), Section 4.5.

The BIS, which is covered in Section 6.8, is able to introduce sufficient negative reactivity into the reactor primary system in order to assure a reactor shutdown from the full power operating condition with no control rod motion.

6.4 Safety Features for Stabilization of the Molten Core

Corium Shield

To mitigate the consequences of a potential Severe Accident (SA), the lower portion of SCCV and RPV pedestal is provided with a corium shield to prevent potential contact between molten core and containment components. The corium shield is discussed in PSR Ch. 9B, Section 9B.2.1.2.1 (Structural Description of SCCV).

6.5 Containment and Associated Systems

The BWRX-300 containment pressure boundary provides a SC1 leak-tight barrier to prevent the release of radioactive material to the environment in the event of a failure of the RCPB. The containment structure maintains its functional integrity during and following peak transient pressures and temperatures that could occur following postulated DBAs. The containment integrity is assured by the isolation of mechanical systems penetrating containment for Postulated Initiating Events (PIEs) that could result in an uncontrolled release of radioactive materials to the environment. Other containment systems support the passive removal of heat generated from AOOs, DBAs, and DECs, and the control of hydrogen and other combustible gases from SA or BDBAs.

6.5.1 Containment Functional Requirements

The PCS design limits are established and confirmed by Deterministic Safety Analysis (DSA) for postulated AOOs and DBAs (refer to PSR Ch. 15, Section 15.5). The PCS provides the FSF of confinement that achieves the following confinement functions:

- Control of pressure and temperature
- Isolation of the confinement boundary
- Leak-tightness of the confinement boundary
- Controlled point of release
- Control of combustible sources
- Reduction of the concentration of free radioactive material in the confinement boundary
- Protection against external events
- Radiation shielding

The BWRX-300 PCS:

- Encloses and supports the NBS, the RPV and connected piping systems
- Provides radiation shielding
- Serves as a boundary for radioactive contamination released from the NBS, or from portions of connected systems located inside the PCS.

6.5.1.1 Energy Management

Post-accident energy management is described in Section 6.5.4 for Passive Containment Cooling System (PCCS). The heat removal capabilities of the PCCS have been analyzed and accepted by the USNRC in NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” (Reference 6-20). The analysis of PCCS heat removal capabilities from normal operation, DBAs, or DECs is addressed in PSR Ch. 15, Section 15.5 for DSA. PCCS heat removal during SA or BDBA is confirmed in PSR Ch. 15, Section 15.6 for Probabilistic Safety Assessment (PSA).

6.5.1.2 Management of Radioactive Material

For AOOs and DBAs, the PCS contains and confines any radioactive materials in accordance with SSR-2/1, Safety of Nuclear Power Plants: Design (Reference 6-13).

6.5.1.3 Management of Combustible Gases

For DECs and BDBAs, the control of combustible gases is analyzed in PSR Ch. 15, Section 15.6 with a description of the combustible gas system in Section 6.5.6.2.

6.5.1.4 Management of Severe Accidents

SA and BDBA management are analyzed and described in PSR Ch. 15, Section 15.6 for PSA.

6.5.2 Primary Containment System

The System Design Description (SDD) for the Primary Containment System (PCS) is 006N7823, "BWRX-300 Primary Containment System (T10) SDD" (Reference 6-6).

6.5.2.1 System and Equipment Functions

PCS encompasses the RPV, attached piping, equipment, and the RIVs, making a leak-tight boundary. The PCS is the fourth fission product barrier (preceded by the fuel pellet, fuel clad, and RPV), and is also a floodable volume to assure core coverage in response to BDBAs. The PCS is a SC1 system that assures the integrity of the barriers prevents releases and maintains the plant in a safe state.

The leak-tight containment vessel confines fission product releases from the core and prevents the release of radioactive contamination to the environment. The PCS outlined in red on Figure 6-6 is a dry, inerted containment with active and passive cooling features that dissipates normal (power operation) and off-normal (transient or accident) heat loads. Table 6-3 provides key parameters used in the developing the BWRX-300 SCCV design.

6.5.2.2 Safety Design Bases

The PCS meets the requirements of SSR-2/1, Safety of Nuclear Power Plants: Design (Reference 6-13):

- Leak-Tight Boundary: Provides a leak-tight boundary in conjunction with other safety class systems/features limiting the radiological effects of contamination released from the NBS and connected systems inside of containment to meet the requirements of SSR-2/1. All boundary components including the SCCV inner wall, containment closure head, equipment and personnel access airlocks, electrical and piping penetrations ensure leak-tight integrity. The PCS establishes the maximum allowable leakage rate under specified parameters for all boundary forming components and provides containment pressure monitoring during all normal and off-normal conditions.
- Structural Integrity: Maintains structural integrity of the pressure boundary preventing structural failures. Structural pressure boundary components including the SCCV, containment closure head, equipment and personnel access airlocks, electrical and piping penetrations, and Containment Isolation Valves (CIVs) maintain structural integrity under all specified normal and off-normal static and dynamic loading conditions, to meet the requirements of SSR-2/1 (refer to PSR Ch. 9, Section 9B.2.1).
- Capability for Pressure Tests: the PCS is subject to pressure testing, at a specified pressure to demonstrate structural integrity. Testing is conducted before plant operation commences, and at appropriate intervals, to meet the requirements of SSR-2/1. Refer to Section 6.5.10.
- Radiation Shielding: Provides permanent radiation shielding within and outside containment limiting doses ALARP. Containment shielding is provided within the RPV core region, and the SCCV structural concrete also functions as radiation shielding external to the containment, to meet the requirements of SSR-2/1.
- Equipment and Personnel Access: Access openings are sized accommodating repair/replacement activities. A removable opening (containment closure head) in the upper containment region allows access to the RPV for refueling outage activities, to meet the requirements of SSR-2/1.

- Structural Support: Provides structural support to safety class SSCs within containment, including the RPV, as well as anchorage for connected piping systems and components, to meet the requirements of SSR-2/1.
- Off-Normal Containment Cooling: Maintains the pressure and temperature within acceptable limits through the PCCS during all off-normal conditions, to meet the requirements of SSR-2/1.
- Leak Detection: Detects leaks within the containment boundary by means of pressure detection and containment sump level detection.

Steel-Plate Composite Containment Vessel Safety Design Basis

The dry, inerted SCCV pressure containing single chamber configuration is designed to accomplish the following safety design basis for structural integrity:

- SCCV safety functions rely on passive design features and natural forces.
- SCCV structure design accommodates the full range of loading conditions consistent with normal plant operation, AOOs and all DBA design loads.
- SCCV structure is sized and equipped to contain the mass and energy released by a large- break LOCA, and for small breaks.
- SCCV structure accommodates the maximum external negative pressure difference relative to the enclosing structure surrounding the SCCV.
- SCCV is protected from or designed to withstand hypothetical missiles from internal and external sources or the uncontrolled motion of broken pipes.
- SCCV pressure boundary wall has an integral or external passive cooling (i.e., PCCS), using air or water, necessary by design analysis and plant layout constraints.
- SCCV forms a fission products release barrier discharged from the RCPB into the SSCV volume as a result of an accident up to and including the defined DBA events.
- SCCV accommodates flooding for long-term core cooling and permits recovery actions up to and including fuel assembly removal following a postulated DBA.

The PCS is designed to accomplish its functions following the initiation of a DBA event. Refer to Table 6-3 for the preliminary primary containment key design parameters.

The containment performance is demonstrated in PSR Ch. 15, Section 15.5 and 15.6. Refer to PSR Ch. 15 for DECs, in accordance with SSR-2/1.

6.5.2.3 Description

The PCS is located in a Seismic Category A below-grade silo and constructed using Steel-Plate Composite modules with diaphragm plates as a SCCV (Figure 6-6) within the RB (Refer to NEDC-34165P, "BWRX-300 UK PSR GDA Ch. 3: Safety Objectives and Design Rules for SSCs," (Reference 6-21), Section 3.5.1). The PCS is part of the nuclear island design and arranged so that the major pools of the nuclear island are above containment. This layout provides protection for PCS and equipment and components from external hazards. This arrangement uses safety class passive cooling methods for the core, RPV and the SCCV.

The PCS design uses a nitrogen-inerted containment atmosphere during most operating modes. Atmosphere control is provided by the Containment Inerting System (CIS) (refer to PSR Ch. 9A, Section 9A.4.2 for the CIS system description). The CIS establishes and maintains an inerted nitrogen atmosphere for the prevention of hydrogen combustion in a post SA condition. The inert atmosphere design minimizes long-term corrosion and degradation of the SCCV and the contained components by limiting oxygen exposure during plant operating service life.

The PCS has provisions for personnel access and habitability during plant outages to perform maintenance, inspections, and tests required to assure SCCV integrity and reliability, and the integrity and performance reliability of interfacing SSCs contained inside the PCS boundary.

The PCS is comprised of the following components:

- Steel-Plate Composite Containment Vessel
- Containment Closure Head
- Personnel and Equipment Airlocks
- Mechanical Penetrations
- Electrical Penetrations
- Refueling Bellows Seal
- Integrated Leak Rate Penetration (refer to Section 6.5.10)
- Containment Corium Shield / Liner (refer to Section 6.4)
- Core Region Biological Shield (Bioshield, refer to PSR Ch. 9B, Section 9B.2.2)
- Passive Containment Cooling System (refer to Section 6.5.3)
- Containment Support Structures
- Maintenance Structures
- RPV Vertical and Lateral Support Structures

Figure 6-6 and Figure 6-7 show a cross sectional views of containment within the reactor building, and Figure 6-8 is a simplified diagram of the containment SSCs.

The sub-compartments in containment are the compartments below the RPV and inside the RPV support structure, the spacing between the RPV and RPV Pedestal, and containment dome above the refueling bellows. Containment sub-compartments do not have large high energy pipes. Therefore, the sub-compartment boundaries are not subject to large transient loads resulting from pipe breaks inside the sub-compartments. Pipe breaks outside the sub-compartments may result in transient loads on the sub-compartment boundaries. However, since the pipe breaks occur in a large volume and because the sub-compartments have sufficiently large openings, these loads are not large in magnitude. The pressure differential across the boundaries is evaluated in Section 6.6.1 of NEDC-33922P-A (Reference 6-20) for the most limiting case and are small. The sub-compartment design includes the resulting differential pressures from pipe breaks outside the sub-compartments.

Steel-Plate Composite Containment Vessel

The SCCV is a concrete and steel structure that is a leak-tight containment boundary, providing structural support and radiation shielding external to containment.

The containment structure is composed of an SCCV, which consists of a Steel-Plate Composite cylindrical wall, basemat, top slab, and steel containment closure head. The containment structure is completely enclosed within the deeply embedded RB that includes containment penetrations and other safety components. The containment structure houses the Steel-Plate Composite internal pedestal that supports the RPV and also provides shielding. The SCCV is also integrated with the reactor building, and the integrated structure is supported by a common Steel-Plate Composite basemat.

The SCCV is classified as a SC1 and Seismic Category A structure. The structure description and analysis comply with all applicable codes and standards provided in PSR Ch. 3, Section 3.5.3. The SCCV structure is discussed in PSR Ch. 9B, Section 9B.2.1.

The SCCV is designed considering the rules and requirements of ASME Boiler & Pressure Vessel Code (BPVC), “Section III, Rules for Construction of Nuclear Facility Components, Division 2, Code for Concrete Containments,” (Reference 6-22).

Containment Closure Head

The containment closure head is a removable steel dome that covers the opening in the top slab of the SCCV that is over the RPV functioning as a portion of the upper containment boundary. The interface between the closure head and the SCCV is a metal ring that is anchored into the top slab of the SCCV and includes a bolted flange for attachment/removal of the closure head.

The containment closure head is removed during reactor refueling and replaced using the RB crane prior to reactor operation.

The containment closure head forms part of the reactor cavity pool and retains water above PCS during normal operation. The containment closure head is SC1 and Seismic Category A structure.

Containment, RPV and Maintenance Support Structures

The internal structural steel is SC1 and Seismic Category A structure that is described in PSR Ch. 9B, Section 9B.2.2.

Containment Support Structures:

The internal structural steel, which includes the Containment Equipment and Piping Support Structures (CEPSS) and support floors at level -21 and -29 m, are steel structures located inside containment (refer to Figure 6-6). The containment internal steel structures are supported by the SCCV wall, RPV pedestal, and the CEPSS which also interface with the RPV at the upper stabilizer locations. The internal structural steel consists of various structural components such as beams and columns.

Refer to PSR Ch. 9B, Section 9B.2.2 for a description of the structural role of the support floors at Level -21 m and -29m, and the primary CEPSS functions.

Maintenance Structures:

Platforms, stairs, ladders, or other structural facilities required for personnel entry and work areas performing inspections, tests, and maintenance or modifications on equipment within containment are provided.

Space is provided around equipment locations inside the SCCV allowing removal, servicing, and maintenance. Platforms and staircases for equipment access allow inspection and maintenance.

Platforms are provided for equipment inspection, examination, surveillance, and maintenance. Platforms and structures do not hinder the SCCV performance and considers the effects of high energy jet and impingement loads minimizing missile and debris generation. Removable stairs and platforms are used in place of permanent installations as needed.

The SCCV has installed crane rail(s) and cart track(s) (typically, monorail or mono-track) and pick points for lifting, positioning, and transporting components, equipment, maintenance tools, materials, inspection and test machines, equipment and tools, servicing systems and components inside the SCCV including the interior side of the SCCV boundary.

RPV Vertical and Lateral Support Structures:

The RPV pedestal is a SC1 and Seismic Category A structure.

The RPV pedestal is a cylindrical-shaped, steel-plate composite structure that structurally supports the RPV. The RPV pedestal, enclosed within the SCCV, is welded to a Steel-Plate Composite basemat that is connected with the SCCV.

Openings are provided in the RPV pedestal to permit pipe routing to the RPV, permit in-service inspection of the RPV and piping, and ensure personnel access into the under-vessel region.

The RPV Pedestal primary safety functions are:

- Provide structural support to SSCs such as the RPV, RPV stabilizers and miscellaneous platforms.
- In conjunction with the Bioshield, the RPV Pedestal provides radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOOs, DBAs, and DECs.

Additional description of the RPV support is provided in PSR Ch. 5, Section 5.9. The primary functions of the Bioshield are described in PSR Ch. 9B, Section 9B.2.2.

Refueling Bellows Seal

The refueling bellows seal is designed as a mechanical appurtenance to the RPV and forms a seal between the outside RPV wall surface and the upper region of the SCCV during refueling operations when the RPV head is removed.

The refueling bellows assembly accommodates the movement of the vessel caused by operating temperature variations and seismic activity. During seismic activity, additional movement is expected and the bellows acts as a buffer or damper so that excessive movement is not transmitted to the RPV from containment.

After operating as a flexible connection during normal plant operation, the refueling bellows must provide a watertight seal between the RPV and containment to support a pool of water over the vessel flange of sufficient depth to allow fuel bundle transfer during refueling. The refueling bellows provides a 360° structural barrier that retains the refueling cavity water above the SCCV when the containment closure head is removed. The refueling bellows design includes protection from puncture or damage from dropped items during refueling outage activities, or workers performing RPV or Containment Closure head removal or installation activities. The design includes a drain to remove water from the bellows low point which can be cleaned (i.e., for removal of non-soluble radioactive contamination, including fuel particles, which settle onto the bellows assembly during refueling outages).

Mechanical Penetrations

Mechanical penetrations function as a portion of the containment boundary consisting of a metal sleeve that is welded into the SCCV and an attached section of process piping, allowing transfer of the process medium across the containment boundary when isolation valves are open. High temperature mechanical penetrations have an air gap between the sleeve and the process pipe. Cold mechanical penetrations have no air gap and the sleeve itself functions as the process pipe.

As indicated above, containment penetrations are outlined in the PCS Simplified Facility Diagram in Figure 6-8. Mechanical penetrations are designed maintaining containment integrity under DBAs conditions, including pressure, temperature, and radiation.

All mechanical penetrations are SC1 and Seismic Category 1A. Piping systems that penetrate containment pressure boundary are equipped with provisions to automatically and reliably seal preventing leakage.

The SCCV is periodically tested to measure the integrated leakage rate from the SCCV structure to confirm the leak-tight integrity of the pressure boundary.

Refer to Figure 6-8 for the PCS Simplified Facility Diagram that includes mechanical penetrations.

Electrical and Instrument Sensing Line Penetrations

Electrical penetrations consist of a metal sleeve which is welded into the SCCV. The penetration assembly forms a leak-tight seal with the containment boundary, while allowing the passage of continuous electrical cables across the boundary.

Instrument fluid sensing line penetrations are arranged in ganged-penetration assemblies.

Integrated Leak Rate Penetration

A dedicated penetration is provided to allow for the pressurization of the containment in order to perform integrated leak rate testing.

Core Region Biological Shield (Bioshield)

The Bioshield is a cylindrical-shaped structure surrounding the RPV pedestal and in conjunction with the RPV Pedestal, provides the radiation shielding during various plant states such as normal operation, AOO, DBA, and DEC. It provides structural support for the containment internal structural steel support floors at Levels -21 m and -29 m.

Shielding is provided inside of containment around the core region of the RPV. Shielding will not extend above connected piping in order to avoid sub-compartment pressurization during postulated line break scenarios.

6.5.2.4 Materials

PCS materials are discussed in PSR Ch. 9B, Section 9B.2.1 in accordance with SSR-2/1, Safety of Nuclear Power Plants: Design, (Reference 6-13).

6.5.2.5 Interfaces with Other Equipment or Systems

Systems interface with the SCCV by directly communicating with the containment atmosphere or penetrating the SCCV pressure boundary for equipment connections located inside the SCCV. See Table 6-4 for PCS interfaces, Figure 6-9 and PSR Ch. 9B, Section 9B.2.1.

6.5.2.6 System and Equipment Operation

The PCS is a passive system. Containment is inerted with nitrogen supplied by the CIS (refer to PSR Ch. 9A, Section 9A.4.2) within 24 hours after the plant reaches 15% reactor power level, in accordance with the OLC. Prior to reactor shut down, containment is supplied with breathable air during the de-inerting process up to 24 hours prior to descending to 15% reactor power level, in accordance with OLC. During normal plant operation, CIS maintains the SCCV slightly above atmospheric pressure conditions preventing non-inert air infiltration. Normal containment cooling is performed by the Containment Cooling System (CCS) (PSR Ch. 9A, Section 9A.5.6).

PCS leakage is monitored using the following parameters:

- Containment sump pump-out rate
- Containment sump level changes
- Air Handling Unit (AHU) drainage flow rates
- Containment atmosphere fission product sampling
- Containment Pressure

The PCS pressure and bulk average and individual sensor temperature readings are continuously monitored with alerts for pressure and temperature above normal range.

The equipment and reactor cavity pool above the containment is used as a heat sink for long-term cooling in abnormal or accident conditions if the CCS fan coolers become unavailable. The PCCS removes heat from the containment and reduces the containment temperature and pressure in abnormal and accident conditions.

Heat transfer occurs from 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.

Fan coolers provide the primary means of heat removal from containment during normal operation (PSR Ch. 9A, Section 9A.5.6 for CCS). Heat is lost to the reactor cavity pool through the containment head.

6.5.2.7 Instrumentation and Control

Instrumentation for interfacing systems is described in Table 6-4. Refer to PSR Ch. 12, Section 12.3.4 for the Containment Monitoring Subsystem (CMon).

6.5.2.8 Monitoring, Inspection, Testing, and Maintenance

Containment monitoring, inspecting, testing and maintenance is described in Section 6.5.10.

Monitoring, inspection, testing, and maintenance are planned and performed to meet system reliability and availability targets.

6.5.2.9 Radiological Aspects

The DSA for the PCS and SCCV are described and analyzed in PSR Ch. 15, Section 15.5.

6.5.2.10 Performance and Safety Evaluation

Containment design pressure and temperature are established with sufficient margin to the calculated peak pressure and temperature for large-break LOCA events. Refer to PSR Ch. 15, Section 15.5.4 for large-break LOCA.

The peak containment pressure and temperature mass and energy release resulting from a large break LOCA are limited by fast closing isolation valves.

The PCS total leakage following a DBA is limited to less than leakage rates that result in offsite doses greater than those set forth by SSR-2/1, Safety of Nuclear Power Plants: Design, (Reference 6-13). Refer to NEDC-34181P, "BWRX-300 UK GDA PSR Ch. 15.3: Safety Analysis – Safety Objectives and Acceptance Criteria," (Reference 6-23), for safety acceptance criteria and NEDC-34185P, "BWRX-300 UK GDA PSR Ch. 15.7: Safety Analysis – Internal Hazards," (Reference 6-24), for the results of both the DSA and PSA compliance with the acceptance criteria.

In the event of pipe failure, the PCS does not have sub-compartments containing large, high-energy pipes preventing dynamic loads from sub-compartments pressurization. The containment shell and the internal structures withstand jet loads resulting from large pipe breaks. The sub-compartments have large openings to the rest of containment preventing a large differential pressure from developing. The pressure differential across the sub-compartment boundaries is evaluated in Section 6.6.1 of NEDC-33922P-A for the most limiting case and found to be small (Reference 6-20). Refer to PSR Ch. 15, Section 15.5 for further analysis.

6.5.3 Secondary Containment System

The BWRX-300 does not employ a secondary or dual containment (see Section 5.3.11 of NEDC-33911P, Revision 3 (Reference 6-14). The BWRX-300 containment system is termed the PCS. The PCS is an ESF that provides a physical barrier against radiological releases to the environment. The PCS is a dry containment type (i.e., unlike many current generation BWRs, the design does not incorporate pressure suppression or a secondary containment). Containment heat removal during accident conditions is provided by the PCCS (see Section 6.5.4). Containment isolation valves, piping, and penetrations are part of the containment boundary.

6.5.4 Passive Containment Cooling System

The System Design Description (SDD) for the PCCS is 006N7777, "BWRX-300 Passive Containment Cooling System (T41) SDD," (Reference 6-25).

6.5.4.1 System and Equipment Functions

During normal operation, PCCS does not contribute to heat removal significantly. Normal containment cooling is performed by the Containment Cooling System (CCS) (PSR Ch. 9A, Section 9A.5.6).

The PCCS maintains temperature and pressure of the Primary Containment System (PCS) through passive cooling during off-normal conditions. The PCCS is based upon proven concepts and simple thermosyphon principles. Heat is rejected to the equipment pool above containment by natural circulation using three independent trains of PCCS pipes as depicted in Figure 6-10. 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. There are no active components or actuation signals required, as the PCCS is always in-service. During normal operation, PCCS does not contribute to heat removal significantly because containment temperature is maintained by the CCS (PSR Ch. 9A, Section 9A.5.6). The PCCS becomes effective when steam is discharged into the containment following a DBA pipe break.

OPEX from the ABWR supported the design development of the PCCS, which relies on natural circulation and condensation to remove heat without needing pumping or instrumentation to control or operate. OPEX also showed that the system should operate passively eliminating operator errors and mechanical failures. The significant reduction in active equipment has reduced the need for intrusive maintenance and operator dose uptake.

6.5.4.2 Safety Design Bases

The PCCS removes heat from the containment and reduces the containment temperature and pressure in accident conditions. The reactor cavity and equipment pools above the containment are used as a heat sink for long-term cooling following DBAs or if the CCS fan coolers become unavailable.

6.5.4.3 Description

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.

Each PCCS train has a containment isolation valve (CIV) and a system isolation valve on the supply and return header outside containment that can be used to manually isolate a train in the event of a PCCS pipe leak or to remove a train from service for maintenance or testing during Cold Shutdown or Refueling Modes when the system is not required. The valves are normally open and are remotely operated from the Main Control Room (MCR). The PCCS containment isolation valves are located outside containment as close to the containment boundary as practical. The PCCS system isolation valves are located outside containment as close to the equipment pool penetration as practical.

The PCCS inlet piping, on each train features a drain valve attached to the piping upstream of the isolation valve. The drain valve enables each PCCS train to be drained to the Liquid Waste

Management System (LWM) for maintenance and testing during the Refueling Mode when the equipment pool is drained.

The PCCS piping outside the containment boundary is open to the equipment pool. The ends of the supply and return pipes in the equipment pool are flanged / threaded with filters ensuring the piping remains free of debris. Each PCCS train can be pressure-tested during the Refueling Mode after the equipment pool and PCCS system is drained.

6.5.4.4 Materials

The materials of PCCS are ASTM A312 / ASME SA312, GR TP316/ TP316L Seamless Stainless Steel (SS) nominal size piping.

6.5.4.5 Interfaces with Other Equipment or Systems

The PCCS is part of the PCS. PCCS interfaces with the equipment pool to maintain containment pressure and temperature within the design limits during accident conditions or loss of active containment cooling. The Fuel Pool Cooling and Cleanup System (FCP) as outlined in Table 6-4 provides water to the equipment pool and heat sink for PCCS. Refer to Figure 6-10 for the PCCS piping layout.

6.5.4.6 System and Equipment Operation

Heat transfer occurs from the containment to the PCCS by natural convection and condensation through the PCCS to the subcooled water in the equipment pool.

The PCCS becomes effective when steam is discharged into the containment following a pipe break. The steam discharged to containment raises containment temperature and increases the steam content for condensation to occur. Condensation heat transfer, even in the presence of non-condensable gas, is much larger than heat transfer from a non-condensable gas at the same temperature. Heat transferred to the PCCS from the containment is removed by the natural circulation of water in single phase flow and rejected to the subcooled water in the reactor cavity equipment pool.

Because the ICs have sufficient capacity to remove decay heat, the majority of the energy during an accident is removed directly from the RPV. PCCS must remove only the energy released from the break to the containment. The energy released from a break is limited for a large break because large breaks are isolated rapidly. Energy released from un-isolated small breaks is also limited due to the small break size. Additionally, the ICs depressurize the RPV making the break flow very small within a few hours. Depressurization of the RPV also reduces the RPV temperature, reducing the heat load from hot surfaces. The PCCS has sufficient capacity to reduce the pressure and temperature in the containment, minimizing leakage following an accident and maintaining pressure and temperature below the design limits in isolation events. Calculations for large and small breaks are presented in PSR Ch. 15, Section 15.5.4.

During Power, Hot Shutdown, and Stable Shutdown Operations, the PCCS is aligned to allow flow of the equipment pool water that is in communication with the reactor cavity pool through the closed loop within containment. The PCCS is not required to be in-service for Cold Shutdown Operation or in Refueling Mode.

6.5.4.7 Instrumentation and Control

There are no C&I inputs for PCCS as a passive system.

6.5.4.8 Monitoring, Inspection, Testing and Maintenance

Monitoring, inspection, testing, and maintenance for PCCS piping are planned and performed to meet system reliability and availability targets. The PCCS pressure test is performed during the modes when PCCS is not required to be operable.

The PCCS is an integral part of the containment boundary and is designed to be periodically pressure tested as part of the overall Containment Leakage Rate Test (Section 6.5.10) program to demonstrate the structural and leak-tight integrity.

6.5.4.9 Radiological Aspects

The DSA for AOOs and DBAs is discussed in PSR Ch. 15, Section 15.5. No design basis AOOs or DBAs lead to core damage. Breaks outside containment are the only DBAs leading to a normal coolant concentration release evaluated in PSR Ch. 15, Section 15.5.9, and this release would not affect the PCCS heat removal performance.

6.5.4.10 Performance and Safety Evaluation

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 or during loss of active containment cooling. The PCCS functions for containment depressurization and heat removal are passive and do not require onsite or offsite electric power system operation. The limiting DBAs for containment pressure and temperature are large-break LOCAs from main steam or feedwater pipe breaks. The PCCS rapidly reduces containment peak pressure following a large-break LOCA. Small un-isolated breaks may become more limiting in the longer term for containment structure temperature. Containment pressure and temperature resulting from AOOs are bounded by those resulting from DBAs. The PCCS removes the energy from containment. Refer to PSR Ch. 15, Section 15.5.4 for further details on DBA LOCA analysis.

The PCCS does not have a high duty function for long-term containment cooling as most of the decay heat from the RPV is removed by the ICS. For RPV isolation and loss of all AC events, containment pressure and temperature are limited by condensation on containment walls and containment heat removal by the PCCS, and by RPV decay heat removal by the ICS. The PCCS rejects heat to the reactor cavity and equipment pools above containment during DBAs as outlined in PSR Ch. 15, Section 15.5 (Reference 6-14).

GOTHIC is used to evaluate the BWRX-300 containment response to mass and energy release from the RPV. The performance of the PCCS is included in the overall containment evaluation method. GOTHIC is a thermal-hydraulics software package that solves the conservation equations for mass, momentum, and energy. GOTHIC has been developed and maintained over 30+ years by NAS Engineering, Inc. It has been used for the design, licensing, safety, and operating analysis of nuclear power plant systems including primary system, containment, and equipment performance. GOTHIC includes the necessary modeling capabilities to represent the piping, reactor vessel, and components of the reactor coolant system.

6.5.5 System for Monitoring Hydrogen and Other Combustible Gases in the Containment

The Process Radiation and Environmental Monitoring System (PREMS) monitors containment atmosphere for oxygen, hydrogen, temperature, pressure, humidity, water level, area radiation, and gross gamma radiation. There is no need to control such gases, as described in the introduction to this chapter.

6.5.6 Mechanical Features of the Containment

6.5.6.1 Containment Isolation System

This section describes the isolation of mechanical systems penetrating primary containment. Containment Isolation Valves (CIVs) are SC1 valves that prevent the uncontrolled release of containment contents in the event of an accident or other conditions by maintaining the integrity of the containment boundary. Piping systems penetrating primary containment are provided with detection, isolation, and containment functions that are redundant, reliable, and proven performance. CIVs are periodically tested to validate operability and determine if valve leakage is within acceptable limits.

Further details with respect to the Containment Isolation Valves are provided within the System Design Description (SDD) for the Primary Containment System (PCS), 006N7823, “BWRX-300 Primary Containment System (T10) SDD” (Reference 6-6).

There are several systems with isolation valves inside containment. The safety analysis assumes that large pipe breaks are quickly isolated by the RIVs following a LOCA preserving reactor coolant inventory before there is any potential for a significant radiological release. For piping that have RIVs, the closest piping terminal end (high stress and fatigue location) to the RPV assembly is located outboard of each set of two in-series RIVs.

Automatic CIVs outside containment are also included even though probability of a pipe break inside containment is small. These CIVs outside containment area are also included for the lines with RIVs. Automatic CIVs outside containment are not required to be fast closing because there is no credible scenario in which fission products greater than what is contained in normal reactor coolant is released to containment. The outside containment automatic CIVs closure time assures containment isolation prior to the first fission product release greater than what is contained in normal reactor coolant in source term evaluations that are completed in the detailed design phase. These closure times are in the order of minutes, as stated in Reference 6-6. Additionally, the valve closing time for all CIVs supports specific break isolation functions balanced with water hammer and valve loading considerations. Small pipes for level instruments use Excess Flow Check Valves (EFCVs).

System and Equipment Functions

The containment isolation function maintains containment integrity by providing protection against the uncontrolled release of radioactive materials from the containment to the environment as the result of an accident.

Lines penetrating containment that provide a potential path for unfiltered radioactive release are automatically isolated by initiating CIVs closure limiting the release of fission products during and after a postulated accident. Redundant CIVs are included in each system line near the containment boundary that close on predefined parameters to prevent releases from containment, and any negative effects from the system in that line. Penetrations are included at the boundary between the line and the containment limiting leakage at the connection. Penetrations are designed to maintain structural integrity during the extreme environmental conditions resulting from a postulated accident and protect against leakage. The CIVs, piping between the CIVs and penetrations, and the penetrations are included as a part of the containment boundary together supporting the limited release of fission product leakage during and after an accident.

Safety Design Bases

All mechanical systems with lines penetrating the containment boundary have a SC function supporting containment integrity by isolating and limiting the release of radioactive leakage to within allowable limits during and following an accident. CIVs are periodically tested confirming operability and determining that valve leakage is within acceptable regulatory limits. Containment penetrations with thermal sleeves have resilient seals and expansion bellows permitting periodic inspection and testing.

The containment isolation design is consistent with the following:

- International Atomic Energy Agency (IAEA) Safety Standards Series No. SSR-2/1, “Safety of Nuclear Power Plants: Design,” (Reference 6-13)
- IAEA Safety Standards Series No. SSG-12, “Licensing Process for Nuclear Installations,” International Atomic Energy Agency, (Reference 6-26)

CIVs in lines penetrating the containment boundary meet regulations and guidance for containment isolation providing protection against uncontrolled releases to the environment and isolating systems that may negatively affect containment. The containment isolation provisions for the ICS steam supply and condensate return lines constitute an appropriate application of the alternative means of meeting SSR-2/1 (Reference 6-13). A single failure does not disable the containment isolation function. The two in-series RIVs that function as CIVs remain open during accident conditions allowing the ICS to function as ECCS.

CIV classification is defined by Safety Class (SC), Seismic Category, Quality Group, and location as described in PSR Ch. 3, Section 3.2. Refer to NEDC-34180P, "BWRX-300 UK GDA PSR Ch. 15.2: Safety Analysis – ID, Categorisation and Grouping of PIEs and Accident Scenarios," (Reference 6-27), for the BWRX-300 Fault Evaluation process. The CIV DL functions are described in the individual fault sequence events in PSR Ch. 15, Section 15.5.

Containment integrity is ensured by containment penetrations, RIVs, CIVs, and the containment piping up to CIVs are designed to Seismic Category A and B per SSR-2/1 (Reference 6-13). The penetrations are designed to ASME Boiler & Pressure Vessel Code (BPVC), "Section III, Rules for Construction of Nuclear Facility Components, Division 1, Subsection NE, Class MC Components," (Reference 6-55). Piping up to CIV's and valves are designed to ASME Boiler & Pressure Vessel Code (BPVC), "Section III, Rules for Construction of Nuclear Facility Components, Division 1, Subsection NCD, Class 2 and Class 3 Components," (Reference 6-12).

Containment Isolation Valves

The following mechanical systems penetrate the containment boundary that require containment isolation:

- Nuclear Boiler System (NBS)
- Process Radiation and Environmental Monitoring System (PREMS)
- Isolation Condenser System (ICS)
- Boron Injection System (BIS)
- Control Rod Drive System (CRD)
- Reactor Water Cleanup System (CUW)
- Condensate and Feedwater Heating System (CFS)
- Chilled Water Equipment (CWE)
- Plant Pneumatic System (PPS)
- Containment Inerting System (CIS)
- Equipment and Floor Drain System (EFS)
- Water, Gas, and Chemical Pads (WGC)
- Passive Containment Cooling System (PCCS)

<u>Mechanical System Interface CIVs</u>

Mechanical system interface for CIVs meet the requirements of the International Atomic Energy Agency (IAEA) Safety Standards Series No. SSG-53, "Design of the Reactor Containment and Associated Systems for Nuclear Power Plants," (Reference 6-29), Section 4.154 (a) and Section 4.154 (b), where applicable.

Nuclear Boiler System:

Two MSLs and a RPV head vent that are included in the RCPB are part of the NBS that penetrates containment. The MSLs CIVs fail in the closed position, with valve actuators that maintain the valves closed by positive mechanical means. This containment isolation is single-failure proof. The main steam CIVs are fast closing and fail-closed type valves.

Process Radiation and Environmental Monitoring System:

The Continuous Hydrogen and Oxygen Sample Panel Train A and B lines penetrate containment and are directly connected to the containment atmosphere.

Isolation Condenser System:

The ICS steam supply and condensate return lines penetrate containment and are a part of the RCPB. Each ICS train (one steam supply line and one condensate return line per train) has two inboard automatic isolation valves.

Vent lines from the ICS tie into MSLs between inboard and outboard CIVs and do not penetrate the containment. Each ICS train vents to the MSL and has two series inboard automatic isolation valves.

Each ICS train provides an interface to the RPV with either BIS or SDC. The interface piping is routed through a containment penetration and includes two in series outboard remote actuated isolation valves. The actual ICS to SDC boundary occurs at the outlet of the most outboard isolation valve for each train (refer to Section 6.2.1 for ICS).

Boron Injection System:

A boron injection line has a remote air-operated outboard CIV and a check valve as the inboard containment and RIV. The injection location is in the ICS “C” loop condensate return line downstream of the two ICS condensate return valves, providing a direct flow path into the reactor. The injection location into the ICS condensate return line requires a separate containment penetration for the BIS injection line (refer to Section 6.8). The air-operated injection valve, when open, provides a flow path to the reactor for injecting a boron solution or demineralized water into the reactor. The injection valve, when closed after an injection event, allows re-establishing containment isolation. Locating a BIS CIV inside containment when needed for beyond design basis accident mitigation is not practical. The ICS RIVs are required to be open for accident mitigation.

Control Rod Drive System:

Hydraulic lines for the FMCRD scram function use penetrations without isolation valves based on the closed system piping outside primary containment and RCPB isolation uses internal ball check valves in the drive design. The CRD system and the associated hydraulic insertion line performs a safety critical function by providing high-pressure water to produce a reactor scram. Each FMCRD includes an integral ball check valve at the drive flange insert port. The check ball operation serves to plug the insert line port and limits the reactor coolant discharged in the event of an insert line break. Additionally, manual isolation valves may be used to further isolate the HCU from the hydraulic insertion line if needed.

Reactor Water Cleanup System:

The CUW mid-vessel suction lines penetrate the containment and are a part of the RCPB. Each CUW mid-vessel suction line has two inboard automatic isolation valves. Both CUW mid-vessel suction lines combine inside containment and have one outboard automatic isolation valve for the combined header. The CUW containment isolation valves close upon receiving an isolation signal from SC1, Safety Class 2

(SC2) and SC3 control platforms. The CUW CIVs fail in the closed position, with valve actuators maintaining the valves closed by positive mechanical means.

Condensate and Feedwater Heating System:

The FW line CIVs fail in the closed position, with valve actuators maintaining the valves closed by positive mechanical means. Each CFS line has two automatic isolation valves inside containment and two automatic isolation valves outside containment.

Chilled Water Equipment:

The CWE supply and return lines penetrate containment and are directly connected to the containment atmosphere if equipment leakage occurs. Each CWE supply line and return line has one inboard and one outboard automatic isolation valve.

Plant Pneumatic System:

Containment PPS includes two distinct sections: one supplies nitrogen and instrumentation air connections to the pneumatic valves inside containment and the other supplies service and breathing air inside containment during plant outages. The nitrogen and service air section are aligned to containment during normal operations. The breathing air section is used during plant outages and isolated during normal operations. Both sections are directly connected to the containment atmosphere. Isolation valve arrangements for these two distinct sections:

- Each CIV for the pneumatic nitrogen and instrumentation air section line has at least one automatic CIV inside and outside containment.
- Each CIV for service and breathing air section lines have one locked closed isolation valve inside containment and one automatic isolation valve outside containment. A simple check valve is not used as the automatic isolation valve outside containment.

Containment Inerting System:

Each CIS line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. Both isolation valves on these lines are located outside the containment vessel to remove them from the harsh containment environment and protect them from the effects of flood and dynamic effects of pipe breaks.

The valves are located as close as practical to the containment vessel. The piping from the containment vessel up to and including both valves is an extension of the primary containment boundary and is designed in accordance with ASME BPVC-III NB-2021 requirements (Reference 6-10). The isolation valves arrangement and connecting piping allows for a single active failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, that does not prevent CIS containment isolation.

Equipment and Floor Drain System:

The EFS line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. There are two CIVs placed in series, located outside containment that are placed as close to the primary containment wall as practical. It is impractical to have an isolation valve inside containment above the floor elevation. The pressurized containment sump tank is credited for containment leak detection. The isolation valves on the EFS containment drain line are normally open to connect the drain line to the pressurized sump. The EFS CIVs close automatically. Upon loss of actuating power, the automatic isolation valves fail closed, providing a greater safety position.

Water, Gas, and Chemical Pads:

The WGC demineralized water supply line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. CIVs include two manual isolation valves with one inboard and one outboard of containment. The WGC System demineralized water manual CIVs are locked closed, only opened during Modes 5 or 6 and are administratively controlled.

Passive Containment Cooling System:

Each PCCS (Section 6.5.3) train has an isolation valve on the supply and return header outside containment to isolate a train in the event of a PCCS pipe leak or remove a train from service for maintenance or testing during Cold Shutdown or Refueling Modes when the system is not required. The isolation valves are normally open and are remotely operated from the MCR, however they will auto-close in the event of a pipe leak. The PCCS isolation valves are located outside containment and as close to the containment boundary as practical. Supply and discharge connections from the pool are connected to closed-loop piping within containment.

6.5.6.2 Systems for Protection Against Over-pressure and Under-pressure

The CIS provides overpressure protection for BDBA. This complementary design feature is discussed in NEDC-34187P, “BWRX-300 UK GDA PSR Ch. 15.9: Safety Analysis – Summary of Results of the Safety Analyses Including Fault Schedule,” (Reference 6-28), Appendix 15D Complementary Defence Line 4 Functions for Mitigating Design Extension Conditions. The containment overpressure vent flow path is a hardened vent in SA cases where containment failure from overpressure may occur.

6.5.6.3 Penetrations

Mechanical Penetrations

Penetrations provide the connection between the containment and penetrating lines to prevent the uncontrolled release of containment contents in the event of an accident or other conditions inside the containment.

Instrument Line Penetrations

Instrument lines penetrating containment are sized, or an orifice installed that ensures that following a line break outside primary containment during normal operation the leakage is minimized. The rate and extent of coolant loss is within the normal reactor coolant makeup capabilities and the integrity and functionality of containment air treatment systems is maintained.

Instrument line penetrations are either single or multiple. The multiple instrument line penetrations have 4, 6, or 8 instrument lines per penetration. Wherever possible, instrument line penetrations are shared minimizing the total quantity of penetrations used. Different systems can share the same penetration but only one system is declared for the penetration designation.

Containment isolation function is applied to all mechanical instrument sensing line penetrations of the SCCV boundary providing the highest reliability for maintaining instrument function while limiting potential radioactive releases if an instrument line is ruptured outside the SCCV boundary.

Penetration Sleeves

Penetration sleeves are used for high energy lines to reduce the effects of high temperature and/or pressure of the penetrating pipe on the containment concrete.

High energy lines are defined for normal plant conditions in system operation where either condition is met:

- Maximum operating temperature exceeds 95°C.
- Maximum operating pressure exceeds 1.9 MPaG. This definition does not apply to accident conditions.

For the BWRX-300 design, each mechanical system line penetration is categorized as either a hot penetration or cold penetration.

Cold penetrations are directly embedded into the SCCV. Hot penetrations are provided with a thermal sleeve that attaches to the SCCV to minimize the conductive heat transfer to the SCCV wall. Penetration sleeves are also used for the CWE lines even though they are not considered high energy lines.

Mechanical system penetrations containing trapped liquid between the CIVs have features that relieve thermally-induced pressurization that comply with IAEA SSG-53, Section 4.160 (Reference 6-29) that states: "Overpressure protection should be provided for closed systems that penetrate the containment and for isolated parts of piping that might be over pressurized by an increase of the temperature inside the containment atmosphere in accident conditions." A self-relieving penetration is typically selected, and the inboard isolation valve is oriented so that excess fluid is released inward to the containment. Using a separate relief valve provides penetration piping overpressure protection is permissible on a case-by-case basis when no other isolation valve selection option is available.

Mechanical system penetrations arrangements provide clearance for inspections.

Containment Isolation Support Piping

The mechanical system piping supporting containment isolation functions for those systems listed in Section 6.5.6.1 (Containment Isolation Valves) is the piping between the inboard and outboard CIVs for most system applications. It also includes the piping from the outboard CIV to their respective Seismic Interface Restraint that is further away from the containment than the valve.

Materials

Containment integrity is ensured by the SCCV penetrations, piping, and valves design to ASME BPVC-III NB-2021 – Class 1 Components (Reference 6-10) with Seismic Category B for valves and Seismic Category A for penetrations and piping (Reference 6-13).

Interfaces with Other Equipment or Systems

Interfacing mechanical systems penetrating the containment boundary that require isolation are discussed in Section 6.5.6.1 (Containment Isolation Valves).

System and Equipment Operation

System and equipment operation is discussed in the following sub-section.

Instrumentation and Control

Containment isolation occurs automatically from closure signals generated by analytical limits or by remote manual. CIV closure is completed once an isolation signal is received. After an isolation valve closes, the valve does not open until the signal is removed, or the operator takes action by resetting the switch. The reactor operator cannot override a containment isolation signal to return the valve to the normal operating position by a single action.

<u>Isolation Signals</u>

Isolation valves that are closed automatically for mechanical system lines that penetrate containment are not capable of being reopened in the presence of an isolation signal. The controls for resetting an isolation signal do not result in the automatic reopening of CIVs.

Position indication for all power operated CIVs are provided in the control rooms. Position indication for power operated CIVs are based on actual valve position and not on demanded valve position. CIVs are configured to permit visual verification of valve position. Check valve CIVs are equipped with closed and open position indication devices, unless justified as not applicable.

Instrument Lines:

Instrument lines penetrating the primary containment are provided with an automatically operated CIV, or one that an operator can manually operate from a remove location, or an excess-flow check valve. Self-actuated excess flow check valves in instrument lines are designed to close when the flowrate increases to a value representative of a loss of piping integrity outside containment.

Self-actuated excess flow check valves in instrument lines have the capability to automatically reopen when the pressure in the instrument line is reduced after previously closing due to an increase flowrate that was representative of a loss of piping integrity outside containment.

<u>Containment Isolation Valves Closure Times</u>

For system lines that have an open path from the containment to the environment, closure times are defined based upon the deterministic safety analysis (PSR Ch. 15, Section 15.5) minimizing the radiological effects in the event of an accident. The outside containment automatic CIV closure times assures containment isolation prior to the first fission product release that is greater than what is contained in normal reactor coolant source term evaluations. For some cases, the closure times are also determined based on minimizing the effects of containment flooding due to a pipe break within containment.

Total closure times consider both the C&I total time to initiate the valve actuation, and the mechanical total time for the valve to full close:

- C&I time – time it takes for the instrument to identify the trip signal to the actual switch that trips the valve actuator.
- Mechanical time – time it takes the actuators to move the valve to fully closed.

The following general criteria are used in establishing closure times:

- Established closure times prevent allowable radiological releases from being exceeded.
- For large breaks, the RIVs close rapidly and prevent significant loss of RPV inventory, uncovering the reactor core, or fuel clad heat-up. The largest diameter steam line and FW pipes are selected for DSA are the most limiting RPV blowdown conditions for plant isolation response.

<u>Valve Position</u>

For the following plant operational modes, CIVs have a typical valve position:

- Normal Operation
- Shutdown
- Post-Accident
- Loss of Motive Power

The positions are typical and based on plant operation and may vary based on system mode changes.

On loss of motive power each CIV takes the position that provides greater safety.

Position Monitoring

The containment isolation function, including provisions for control, indication, and performance under loss/restoration of power conditions, is instrumental in maintaining containment barrier integrity and not interfering with flow paths essential for reactor core cooling.

Position indication for all power-operated CIVs are provided in the MCR. Position indication for power-operated CIVs are based on actual valve position and not on demanded valve position. CIVs are configured to permit visual verification of valve position.

Manually operated valves are locked closed and are administratively controlled.

Allowable Leakage

The BWRX-300 CIVs minimize leakage. This section defines requirements specific to leakage performance. Allowable leakage for containment leak rate testing is provided before preoperational testing. A discussion on containment leakage testing is included in Section 6.5.10.

System Specific Allowable Leakage Requirements

Leak rates are determined by manufacturer accepted seat leakage test methods for each systems specific design parameters.

General Allowable Leakage Requirements

Test requirements are determined for preoperational and periodic verification of primary containment SSCs to confirm leak-tight integrity and establishing test acceptance criteria. These tests are performed to assure that (a) leakage through the primary reactor containment and systems and components does not exceed allowable leakage rates values and (b) periodic reactor containment penetrations and isolation valve surveillance is performed so that any necessary maintenance and repairs are identified.

Monitoring, Inspection, Testing and Maintenance

All CIVs have individual leakage tests that are performed supporting verification that containment leakage is within the allowable limits. All CIVs are leak tested as discussed in Section 6.5.10, containment leakage testing in accordance with SSR-2/1. Design features required to support testing are defined within this section.

To support containment leakage testing, CIVs are manually cycled in the same manner that the valve closes upon receipt of an isolation signal. For example, if a valve closes automatically by pneumatic operation from an isolation signal, then the valve is also closed remotely by manual pneumatic operation.

Mechanical pipelines penetrating containment contain test connections that support CIV leakage testing. This test is performed by applying a test pressure through one test connection in the same direction the valve performs its safety function. A second test connection is located on the opposite side of the valve for line venting.

Radiological Aspects

Penetrations are arranged to provide sufficient space and shielding ensuring planned maintenance and operations is performed minimizing personnel exposure.

Performance and Safety Evaluation

The performance requirements (time to closure) for CIVs are confirmed by the deterministic safety analysis in PSR Ch. 15, Section 15.5.

6.5.6.4 Airlocks, Doors and Hatches

Personnel and Equipment Airlocks

Two containment airlocks function as a portion of the containment boundary when closed, one in the upper elevation and one in the lower elevation. The containment airlocks are sized to accommodate removal of the most limiting components.

6.5.7 Annulus Ventilation System

This section is not applicable to the BWRX-300 design.

6.5.8 Ventilation System

Control Building Heating, Ventilation and Air-Conditioning System is discussed within PSR Ch. 9A, Section 9A.5.2.

6.5.9 Filtered Venting System

Refer to PSR Ch. 15.9, Appendix 15D for the filtered venting system, which is to be further analyzed to determine if filtered venting is required for BDBA.

6.5.10 Containment Leakage Testing

Containment Leakage Rate Testing is performed to assure leakage through containment and systems and components penetrating the SCCV does not exceed allowable leakage rate values as specified in plant OLC and associated bases. The testing measures the rate at which a contained air mass escapes through the containment boundary at a specific pressure using sensitive instrumentation as described in 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors,” (Reference 6-30), and ANSI/ANS-56.8, “Containment System Leakage Testing Requirements,” (Reference 6-31). Additionally, periodic surveillance of SCCV penetrations and isolation valves is performed to verify that proper maintenance and repairs are made during the service life of the SCCV.

The Integrated Leak Rate Test (ILRT) consists of the containment leakage rate and Local Leakage Rate Testing (LLRT). Testing requirements are in accordance with 10 CFR 50, Appendix J (Reference 6-30), and ANSI/ANS 56.8, Section 3.2.3 (Reference 6-31).

This testing is required on all boundaries that serve as barriers to the release of primary containment atmosphere following a design basis LOCA as identified in 10 CFR 50 Appendix J III.D.2(b)(i) and 10 CFR 50 Appendix J III.C.1 (Reference 6-30).

Testing is performed prior to placing containment into service and periodically during plant operation, as part of the in-service testing examination and testing program ANSI/ANS-56.8-2020, Section 3.2.2 (Reference 6-31), and 10 CFR 50 Appendix J III.D.1 (Reference 6-30), respectively.

Preoperational Leakage Rate Testing

The preoperational ILRT determines whether SCCV structures comply with specified strength and design requirements in accordance with the requirements of ANSI/ANS-56.8, Section 3.2.2 (Reference 6-31). The preoperational proof and leakage rate testing is performed upon completion of construction and prior to criticality in accordance with the requirements of ANSI/ANS-56.8, Section 3.2.2 (Reference 6-31).

The preoperational ILRT includes general visual inspection of the accessible interior and exterior surfaces of the SCCV and components prior pressurization. The visual inspection is performed to identify deterioration that may affect leak tightness in accordance with ANSI/ANS-56.8-2020, Section 3.2.1 (Reference 6-31).

Periodic ILRT

Periodic ILRTs is performed by scheduled test frequency after the SCCV is placed into service. The test includes visual inspection of SCCV as described above. The test includes a positive pressure test of the containment and penetrations. Results of visual examination and ILRT are reported according to the requirements in ANSI/ANS-56.8-2020, Section 5.11.1 (Reference 6-31).

Periodic ILRT Requirements

Containment integrity is verified through preoperational and periodic leak rate testing using established acceptance criteria as identified in ANSI/ANS-56.8 (Reference 6-31) and USNRC 10 CFR 50, Appendix J (Reference 6-30). Testing requirements includes:

- Leakage test requirements
- Test instrumentation
- Test procedures
- Test methods
- Acceptance criteria
- Data analysis
- Inspection and recording test results
- Guidance on required component and pathway testing
- Test frequency

LLRT confirms the SCCV leak-tight integrity boundary in accordance with:

- IAEA Safety Standards Series No. SSR-2/1, “Safety of Nuclear Power Plants: Design" (Reference 6-13)
- USNRC 10 CFR 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors (Reference 6-30)
- ANSI/ANS-56.8, Containment System Leakage Testing Requirements (Reference 6-31)

ILRT procedures are written to satisfy the test method guidance of ANSI/ANS-56.8, (Reference 6-31) and is consistent with USNRC 10 CFR 50, Appendix J (Reference 6-30). The procedures are used to measure leakage rate of the SCCV boundary and evaluate the results to ensure the maximum allowable containment leakage is not exceeded. Refer to PSR Ch. 3, Section 3.10 for a discussion on in-service monitoring, tests, maintenance, and inspections.

ILRT Examination Methods

The ILRT test method is the absolute method of leakage determination using the mass plot analysis technique for containment ILRT in accordance with ANSI/ANS-56.8, Section 5.6.2 (Reference 6-31). The tests are to confirm that the actual containment leak rate does not exceed the maximum allowable leakage at pressure used in the plant safety analysis to demonstrate that release limits are not exceeded, as per ANSI/ANS-56.8, Section 6.4 (Reference 6-31).

Results of the containment ILRT are validated by the performance of a verification test as per the guidance ANSI/ANS-56.8, Section 5.9.4 (Reference 6-31). This may be achieved by one of the following methods:

A. A superimposed known leakage rate.

B. The use of two independent measurement systems to monitor and collect leakage rate data.

Verification of test acceptance criteria is performed in accordance with ANSI/ANS-56.8, Section 5.9.4 (Reference 6-31).

Following a controlled pressure change, the pressure and temperature is allowed to stabilize in accordance with ANSI/ANS-56.8, Section 5.9.3 (Reference 6-31).

Duration of ILRT

ILRT duration is determined by the ILRT lead based on the allowable leakage rate and the accuracy of measurements established for each test procedure in accordance with ANSI/ANS-56.8, Appendices (Reference 6-31).

Isolation, Repair, or Adjustment to Leakage Path

Any modification component replacement or repairs or adjustment which could affect SCCV integrity requires appropriate testing to demonstrate that the affected components meet the applicable leakage requirements ANSI/ANS-56.8, Section 2 (Reference 6-31). These include:

- Any requirements for closure of all containment valves, doors, and hatches.
- Requirements for any test boundary isolation that differs from normal operation.

Pressurized Components

The primary containment is pressurized at a safe rate with air that is clean, relatively dry, and free of contaminants in accordance with ANSI/ANS-56.8, Section 5.5 (Reference 6-31).

Sources of compressed gas are isolated or removed otherwise a compressed gas leakage rate test phase is conducted to ensure that valid leakage rates are being measured in accordance with ANSI/ANS-56.8, Section 3.2.6 (Reference 6-31).

Containment Isolation Valve Closure

Closure of penetration isolation valves is accomplished by normal means and without any preliminary exercising or adjustment. Exercising valves for the purpose of improving leakage performance is not permitted. Repairs of malfunctioning or leaking valves shall be made as necessary as per the guidance in ANSI/ANS-56.8, Section 3.3.3 (Reference 6-31).

Acceptance Criteria

Acceptance criteria ensures that the measured containment boundary leakage rate does not exceed maximum allowable containment leakage at test pressure. This limit demonstrates that release limits are not exceeded, in accordance with ANSI/ANS-56.8, Section 5.8 (Reference 6-31).

Reporting of Periodic ILRT

Reports documenting the results of each preoperational and periodic tests includes a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable. Additionally, for the periodic test, the report shall include an analysis and interpretation of the ILRT and a summary analysis of LLRTs that were performed since the last ILRT in accordance with 10 CFR 50 Appendix J V.B.1 and 10 CFR 50 Appendix J V.B.2 (Reference 6-30).

6.6 Habitability Systems

6.6.1 Control Room Habitability

The BWRX-300 design does not consider the MCR habitability a SC1 function. It is not a SC1 function because there are no control room operator actions credited in the conservative design basis analysis. The Secondary Control Room (SCR) is physically and electrically separate from the MCR as required by SSR-2/1 (Reference 6-13). However, in accordance with SSR-2/1, a physically and electrically separate SCR is provided in the RB when the MCR becomes uninhabitable. Events that necessitate evacuation to the SCR include fire, malevolent acts, and aircraft impact, as example.

The MCR located in the Control Building (CB) and the SCR located in the RB provide environments to protect the operators during all operational states and maintain the reactor in a safe condition. No DBA and DEC can simultaneously affect both control rooms to the extent that the FSFs cannot be performed as required by SSR-2/1 (Reference 6-13).

Control Room Habitability (CRH) refers to the conditions required for life support and safe, effective operation of the plant during operational states and accident conditions. CRH is served by a combination of individual systems that collectively provide habitability. These systems are listed below:

- Heating Ventilation and Cooling System (HVS) (see 006N7781, “BWRX-300 Heating Ventilation and Cooling System (U41) SDD,” (Reference 6-33), for further details)
- Process Radiation and Environmental Monitoring System (see 006N7938, “BWRX-300 Process & Radiation Monitoring System (D11) SDD,” (Reference 6-34), for further details)
- Lighting System, R15 Lighting and Service Power System SDD (007N3201)
- Fire Protection System (FPS) (see 006N7785, “BWRX-300 Fire Protection System (U43) SDD,” (Reference 6-35), for further details)

Habitability features include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, lighting, and fire protection.

The design bases and descriptions of the various habitability features are contained in the following sections:

• Evaluation of Site-Specific Hazards	Section 2.2
• Protection Against External Hazards	Section 3.3
• Protection Against Internal Hazards	Section 3.4
• Control Room Habitability	Section 6.6
• Control Building Heating, Ventilation and Air-Conditioning System	Section 9A.5.2
• Fire Protection System	Section 9A.6
• Lighting and Emergency Lighting Systems	Section 9A.9.2
• Process Radiation Monitoring (PRM)	Section 11.5
• Design for Radiation Protection	Section 12.3
• Shielding	Section 12.4
• Area Radiation Monitoring (ARM)	Section 12.3.4

The MCR and SCR instrumentation is described in PSR Ch. 7, Section 7.5, and Section 7.6, respectively. The CB structural evaluation for external hazards is provided in PSR Ch. 3, Section 3.3 Protection Against External Hazards. Refer to PSR Ch. 9B, Section 9B.3.2 for the CB structural integrity, which houses the MCR. There are no fission product releases resulting from a DBA or AOO as described in PSR Ch. 15, Section 15.5.

In addition to the MCR and SCR, the BWRX-300 design incorporates emergency support facilities as required by SSR-2/1 (Reference 6-13) and described in NEDC-34191P, "BWRX-300 UK GDA PSR Ch. 19: Emergency Preparedness and Response," (Reference 6-32).

6.6.1.1 System and Equipment Functions

The fundamental design philosophy is that the plant is operated from the MCR and only evacuated if necessary. For most PIEs, the operators remain in the MCR to safely operate the plant. If the MCR becomes uninhabitable, is expected to become uninhabitable, or functionality is unacceptably impaired, plant control is shifted from the MCR to the SCR. SCR habitability ensures the plant can be maintained in a safe shutdown condition. The facility cannot be operated at power from the SCR.

A qualified route exists from the MCR to the SCR for events which necessitate evacuation of the MCR. An alternate route exists for fire events. MCR and SCR habitability provisions are provided to ensure that continued occupancy in one of the two locations is possible under PIEs for a minimum of 72 hours as required by SSR-2/1 (Reference 6-13).

Refer to NEDC-34164P, "BWRX-300 UK GDA PSR Ch. 2: Site Characteristics," (Reference 6-36), Section 2.2 for the evaluation of site-specific hazards which includes toxic gas assessments. The MCR is the assured operating location for smoke/external fire and toxic gas release event.

Both the MCR and SCR are protected for internal and external hazards and radiological events. The control room design features are based on proven technologies and human factors engineering considerations that are described in NEDC-34190P, "BWRX-300 UK GDA PSR Ch. 18: Human Factors Engineering," (Reference 6-37), Section 18.4. A description of equipment to support CRH in the MCR and SCR is detailed below.

<u>Main Control Room</u>

As indicated above, the MCR, located in the CB, is the primary location for plant operators.

The CB is normally air conditioned and heated by 2x100% capacity chilled water supplied AHUs located on the CB roof, discharging through supply ductwork and a return plenum.

To accommodate off-normal MCR habitability contingencies, the following air handling units are provided for the CB:

- CB Toxic Gas Filtration Units (TGFUs) operate automatically when toxic gas is detected at the CB AHU outside air intakes. In this event, normal outside air supply to the operating CB AHU isolates and the TGFU discharge damper opens, allowing the associated TGFU to supply pressurization air to the CB through the normal CB supply AHU that continues to operate during a toxic gas event.
- Main Control Room Envelope (CRE) Emergency Filter Units (EFU) operate automatically upon detection of high radiation level at the operating CB supply AHU outside air intake. The CRE isolation mode initiates when normal CB supply and return air dampers close. The operating CB normal supply AHU intake de-energizes upon high radiation detection and auto-start of the standby unit is defeated. Battery Room exhaust fans continue to operate based on timers.

Secondary Control Room

The SCR, located in the RB, is the assured shutdown location for the plant if habitability and control from the MCR is lost.

The SCR performs the following functions when the MCR is inhabitable:

- Initiate shutdown of the reactor and maintain the plant in a safe shutdown condition.
- Monitor FSFs

Emergency Filtration Units:

The SCR is provided with 2x100% EFUs and pressurization fans that supply ventilation air to the operators when automatically placed into service upon radiation, or smoke detection in the normal supply duct. A loss of power to both normal supply AHUs initiates operation of the SCR EFUs and pressurization fans. These two pressurization fans draw outside air through dedicated ducting with blast resistant openings.

Normal Ventilation:

The SCR is normally provided filtered, conditioned air from the operating RB Lower-Level AHU. Normal Ventilation is separate from the emergency ventilation and ensures that the SCR temperature is maintained in a specified range. Electric duct heaters assist in maintaining SCR temperature in a normal range.

Isolation Dampers:

The SCR makeup air handler intake ducts have isolation dampers that are closed on high radiation, smoke, or toxic gas to protect personnel from these hazards.

6.6.1.2 Safety Design Bases

Habitability provisions ensure that continued occupancy in one of the two locations remains possible under PIE as required by SSR-2/1 (Reference 6-13). The design provides provisions for both internal and external events that pose a direct threat to continued operation of the MCR and SCR, and practicable measures to minimize the effects of AOOs, DBAs, or DECs.

MCR and SCR habitability provisions ensure that continued occupancy in one of the two locations is possible under PIEs for a minimum of 72 hours as required by SSR-2/1 (Reference 6-13).

The following regulations, standards and guidance are applicable to the BWRX-300 MCR and SCR design:

- IAEA Safety Standards Series No. SSG-12, “Licensing Process for Nuclear Installations,” (Reference 6-26)
- IAEA Safety Standards Series No. SSR-2/1, “Safety of Nuclear Power Plants: Design,” (Reference 6-13)
- 006N9004, “BWRX-300 Deterministic Safety Analysis Performance Requirements,” (Reference 6-38)
- 006N7785, “BWRX-300 Fire Protection System (FPS) - U43,” (Reference 6-35)

Main Control Room and Secondary Control Room Design Basis

The BWRX-300 design provides an MCR, from which the plant can be safely operated as required by SSR-2/1 (Reference 6-13). The design provides an SCR, from which the plant can be placed and kept in a safe-shutdown state, when habitability is lost in the MCR, as required by SSR-2/1 (Reference 6-13):

- Radiation exposure in the MCR, and SCR when applicable, to personnel throughout the duration of the postulated DBA does not exceed the dose acceptance criteria of 006N9004, “BWRX-300 Deterministic Safety Analysis Performance Requirements” (Reference 6-38).
- The MCR and SCR habitability design features detect and protect personnel from smoke and airborne radioactivity. The MCR habitability design features additionally detect and protects personnel from toxic gas.
- The MCR and SCR habitability requirements are satisfied without the need for individual breathing apparatus or special protective clothing.

Because the BWRX-300 design does not consider control room habitability a SC1 function, the EFUs are not SC1 and are therefore not credited for mitigation of DBAs.

6.6.1.3 Description

The habitability aspects of MCR and SCR HVAC is discussed in Section 6.6. Descriptions of the control room HVS, FPS and Lighting System are found in PSR Ch. 9A, Sections 9A.5.2, 9A.6 and 9A.9.2, respectively. Descriptions of the ARM and PRM are found in PSR Ch. 11, Section 11.5, and PSR Ch. 12, Section 12.3.4. Refer to PSR Ch. 9A, Figure 9A.5-1 for the RB HVAC Process Flow Diagram.

<u>Main Control Room Description</u>

The layout of the MCR habitability area in the CB is shown on Figure 6-11 to Figure 6-13.

CRE isolation and CRE EFU operation provides a pressurized envelope relative to adjacent spaces, maintaining CRE habitability during an airborne radiation release event.

The Main CRE is the area maintained for habitability purposes. Figure 6-11 depicts the Main CRE.

Emergency Filtration Units:

Main CRE EFUs operate automatically upon high radiation level detection at the operating CB supply AHU outside air intake. The CRE initiates isolation mode when normal CRE supply and return air dampers close. Automatic operation of the CRE EFUs provide suitable atmosphere for habitability and maintains the CRE slightly pressurized. The AHU de-energizes upon high radiation detection at the intake and auto-start of the standby unit is defeated.

Radiation Protection:

The CRE isolation and EFU operation provides a pressurized envelope relative to adjacent spaces, maintaining CRE habitability during a radiological release.

Fire Protection and Toxic Gas:

The normal outside air intakes are monitored for toxic gases and smoke and isolates the outside air dampers if toxic gas or smoke is detected.

<u>Secondary Control Room Description</u>

The SCR, located in the RB (refer to Figure 6-6) remains habitable for at least 72 hours following an event requiring activation of the SCR.

The layout of the SCR is shown in Figure 6-14.

SCR Emergency Filtration Units:

The EFUs outside air supply portion of the control room HVS is SC2 and Seismic Category B. Single active failure protection is provided by using two trains that are physically and electrically redundant and separated. In the event of one train failure, the failed train is automatically isolated, and the alternate train is automatically initiated. Each 100% capacity train is capable of supplying filtered air to the SCR pressure boundary at the required flow rate.

A small amount of leakage out through the SCR pressure boundary is anticipated through door seals, electrical penetrations, and isolation dampers. The exhaust from the SCR is optimized to ensure proper scavenging of air from the SCR in an amount equal to the supply. Backflow prevention through the controlled leak path is not required because the SCR is at positive pressure during normal and emergency operation.

The SCR EFU utilizes a High Efficiency Particulate Air (HEPA) filter, carbon filter, and HEPA post filter to provide radiological protection of the SCR outside air supply. The units, along with the associated intakes, dampers, and ductwork are SC2, Seismic Category B, and meet the requirements of 006N7785, “BWRX-300 Fire Protection System (FPS) - U43” (Reference 6-35). Safety class equipment such as the SCR EFUs are qualified to operate in the harsh environment that will be encountered at specific locations following a DBA or DEC. Each of the two SCR EFU trains incorporate a 100% capacity pressurization fan powered by separate division power supply. The SCR EFU trains ensure adequate fresh air is delivered and mixed in the SCR habitability area.

SCR Pressurization Fans:

The SCR is provided with EFUs and pressurization fans that supply ventilation air to the operators when automatically placed into service upon detection of radiation or smoke at the normal lower-level supply AHUs. A loss of power to both normal supply AHUs will also initiate operation of the SCR EFUs and pressurization fans.

Isolation Dampers:

The SCR habitability area pressure boundary includes penetrations, dampers, or valves, interconnecting duct, and related test connections and manual valves. The isolation dampers are classified as SC3 and Seismic Category B. If radiation is detected, the SCR normal supply and exhaust isolation dampers will shut, the SCR pressurization fans will energize supplying the SCR with filtered outside air.

Tornado dampers close automatically and mitigate the effect of external hazards, as required by SSR-2/1 (Reference 6-13).

Radiation Protection:

SCR instrumentation for monitoring of radioactivity is provided in PSR Ch. 7, Section 7.6, and PSR Ch. 12, Section 12.3.4.

Shielding Design:

Shielding design bases is provided in PSR Ch. 12, Section 12.3.2. Descriptions of the design bases source terms and shielding parameters are presented in PSR Ch. 12, Section 12.3.2 and PSR Ch. 15.7, respectively.

Fire Protection:

Fire and smoke detectors, and smoke removal descriptions are provided in PSR Ch. 9A, Section 9A.6.

6.6.1.4 Materials

Refer to PSR Ch. 9A, Section 9A.5.1 for information pertaining to HVS materials.

6.6.1.5 Interfaces with Other Equipment or Systems

Interfaces are addressed in PSR Ch. 9A Table 9A.5-1, Reactor Building System Interfaces and Table 9A.5-2, Control Building Heating, Ventilation and Air-Conditioning System Interfaces.

6.6.1.6 System and Equipment Operation

SCR and MCR emergency habitability control room HVS (PSR Ch. 9A, Section 9A.5.2) do not operate during normal conditions. The control room HVS maintains the air temperature of the control room habitability envelope within a predetermined temperature range using fan cooling units and air handling units that are supplied with chilled water during normal operations and shutdown. This maintains the SCR emergency habitability function passive heat sink at or below a predetermined temperature.

<u>Main Control Room</u>

Figure 6-11 and Figure 6-12 show the MCR habitability envelope and location of AHUs, toxic gas monitors, TGFUs, and CRE EFUs located in the CB.

High Radiation:

CRE EFUs operate automatically upon detection of high radiation level at the operating CB supply AHU outside air intake. The CRE goes into isolation mode when normal CRE supply and return air dampers close.

Loss of All Alternating Current:

During a loss of all AC, the MCR remains the control location until habitability temperatures are exceeded or controls and indications are lost due to high temperature effects from C&I and electrical components. The control room operators will relocate to the SCR in the RB if the MCR becomes uninhabitable.

<u>Secondary Control Room</u>

Operation of the emergency habitability portion of the SCR HVAC is automatically initiated on any of the following conditions:

- High radioactivity in the operating RB normal supply AHU outside air intake.
- Toxic gas detection in supply duct to the RB.
- Loss of all AC power.

Operation is also initiated manually (PSR Ch. 9A, Figure 9A.5-1 for RB HVAC process flow diagram).

High Radiation:

Upon receipt of a high radiation in the operating RB normal supply AHU outside air intake, the normal RB supply and exhaust are isolated from the control room habitability envelope by automatically closing the isolation dampers in the system ductwork. Simultaneously, one of two emergency pressurization fan EFUs automatically starts and begins to deliver filtered air from one of the two unique safety category outside air intake locations.

Loss of all Alternating Current:

The SCR remains habitable for 72 hours during a loss of all AC. During loss of all AC power, AC power is provided from SC1 batteries.

6.6.1.7 Instrumentation and Control

Instrumentation required for SCR habitability design features are provided in PSR Ch. 7, Section 7.6.

6.6.1.8 Monitoring, Inspection, Testing and Maintenance

Monitoring, inspection, testing, and maintenance are planned and performed to meet system reliability and availability targets. PSR Ch. 9A, Section 9A.5.1 describes HVS inspection and testing.

6.6.1.9 Radiological Aspects

The radiological dose consequences from DBAs are analyzed and evaluated, and the accompanying results are provided in PSR Ch. 15, Section 15.5 and results are in Section 15.7.

6.6.1.10 Performance and Safety Evaluation

Radiation doses to SCR and MCR personnel are calculated for the accident scenarios where the pressurization system provides filtered air pressurizing the CRH envelope. Radiological dose consequences from normal coolant pipe break outside containment are evaluated in PSR Ch. 15, Section 15.5. There are no DBAs or AOOs that lead to fission product releases.

6.7 Systems for the Removal and Control of Fission Products

Engineered Safety Feature Atmosphere Cleanup Systems

The design does not require an ESF atmospheric cleanup system. As discussed in Section 5.1.14 of NEDC-33911P (Reference 6-14), the BWRX-300 low-leakage containment is constructed in the subterranean, which is expected to further limit potential fission product leakage.

Containment Spray Systems

This section is not applicable to the BWRX-300 design as addressed in Section 5.3.10 of NEDC-33911P (Reference 6-14).

Fission Product Control Systems

As stated in Section 5.3.11 of NEDC-33911P (Reference 6-14), the design does not use a dual or secondary containment. Containment isolation and leak rate testing is addressed in Section 6.5.10.

Ice Condenser as a Fission Product Cleanup System

This section is not applicable to the BWRX-300 design as addressed in Section 5.3.4 of NEDC-33911P (Reference 6-14).

Pressure Suppression Pool as a Fission Product Cleanup System

This section is not applicable to the BWRX-300 design as addressed in Section 5.3.4 of NEDC-33911P (Reference 6-14).

6.8 Other Engineered Safety Features

This section will present relevant information on any other engineered safety features implemented in the plant design that are not covered in any previous sections.

6.8.1 Corium Shield

The Corium Shield is discussed within Section 6.4.

6.8.2 Boron Injection System

The System Design Description (SDD) for the BIS is 006N7417, "BWRX-300 Boron Injection System (G11) SDD," (Reference 6-39).

The BIS provides a separate, diverse means, Defence-in-Depth (D-in-D) backup system to for manually inserting negative reactivity into the reactor core. The BIS assures reactor shutdown by mixing a neutron absorber with the primary coolant.

The BIS is only required in the highly improbable event when shutdown using CRDs via hydraulic or motor run-in cannot be accomplished. The system introduces sufficient negative reactivity into the reactor primary system assuring reactor shutdown to a subcritical state with no control rod motion.

Injection of a neutron absorber solution for reactivity control is a Defence Line 4b (DL4b), Safety Category 3 function.

The system achieves cold shutdown conditions with a predetermined mass of water in the reactor vessel using an enriched Boron-10 solution injected into the reactor vessel. The CRD system and the BIS do not share any components.

6.8.2.1 System and Equipment Functions

The BIS simplified flow diagram of the boron injection mode is shown on Figure 6-16. The BIS consists of a storage tank, test tank, injection pump, piping, valves, and instrumentation and controls necessary to prepare and inject the neutron absorbing solution into the reactor and for system testing. The air-operated injection valve and the Class 1 injection piping are part of the BIS. The BIS consists of a single, 100% equipment train located in the RB outside containment, tying into an ICS return line between the redundant actuation valves and the RIVs.

The neutron absorber is pumped from the storage tank through the injection valve into the reactor vessel by remote manual operation of the pump, storage tank outlet valve, and injection valve.

The BIS injects the neutron absorber solution into the reactor core compensating for various reactivity effects that could occur during plant operation.

The neutron absorber is an aqueous solution of decahydrate ($Na_2B_{10}O_{16}10H_2O$). Enriched sodium pentaborate solution is made by mixing granular enriched sodium pentaborate with water. The neutron absorber design concentration has a 4.4 °C solution precipitation temperature. This low precipitation temperature eliminates the need for storage tank heating during normal standby conditions or heat tracing the instrument lines and pump suction piping.

The volume versus concentration limits is calculated accounting for normal reactor vessel water volume and water volume in the shutdown cooling piping. This neutron absorber solution quantity is the amount above the pump suction shutoff level in the storage tank. No credit is taken for the portion of the storage tank volume that cannot be injected.

6.8.2.2 Safety Design Bases

The BIS provides a means of achieving cold subcriticality by mixing a neutron absorber with the primary coolant. This condition creates a design requirement for reactivity control from full power to cold 20 °C (68 °F) subcritical state using an enriched Boron-10 solution.

The ICS for overpressure control and CRD/HCU/FMCRD for reactivity control address failure-to-scram impacts or other reactivity events where boron injection provides an additional layer of defence.

The neutron absorber injection is manually initiated by the operator from either the MCR or SCR.

The minimum amount of neutron absorber required to shut down the reactor is calculated based upon the minimum natural boron concentration required for shutdown and the weight of the water in the RPV at normal level including the shutdown cooling loops at cold shutdown conditions. This ensures that total quantity of stored neutron absorber is sufficient to achieve the minimum sodium pentaborate concentration required for shutdown.

The BIS is a complimentary design feature that provides D-in-D for long-term reactivity control DEC.

The BIS system is designated as SC3.

The BIS can be operated in the event of a LOPP but does not perform any non-safety or safety class functions during off-normal conditions.

The BIS injection line penetrates containment and is directly connected to the reactor vessel through the ICS. This portion of the BIS through the ICS system that requires containment isolation is part of the RCPB and is classified as DL3, Quality Group A, Safety Category 1, and Seismic Category 1-A and 1-B. The ICS equipment is SC1.

6.8.2.3 Description

The BIS is a single 100% capacity system that includes:

- A single triplex injection pump capable of 2.27 m^3/hr (10 gpm) pumping against the maximum reactor pressure of 12.41 MPaG (1800 psig)
- A test tank and associated piping and valves
- An injection line with containment isolation valves
- A demineralized water connection
- A service air connection
- System piping drains and collection subsystem

The above equipment is located outside primary containment allowing access for testing and inspection activities during all plant operating conditions. See Figure 6-15 for a Simplified Flow Diagram of the BIS.

Operation in the injection mode requires BIS pump start and opening the storage tank outlet valve and injection valves. The storage tank outlet valve and injection valve are air-operated. The air-operated valves open on a system initiation signal, and close [fail as-is] on loss or removal of the pressurized air source or close on a manual valve closure signal.

The injection location is the condensate return line of the ICS “C” loop downstream of the two ICS condensate return valves, providing a direct flow path into the reactor. The injection location requires a separate containment penetration for the BIS injection line. Overpressure protection of the BIS is provided by a relief valve located on the pump discharge piping. To avoid water hammer at the onset of BIS injection, the piping is fully-filled by a connection with the demineralized water system.

The testing subsystem consists of a test tank containing demineralized water and the associated valves and piping in parallel with the normal suction flow path from the storage tank. During the non-testing phase of system normal operation, the testing network is isolated from the balance of the system by shutoff valves. Periodic operational tests of the BIS are conducted at any time regardless of the state of reactor operation. The testing subsystem demonstrates:

- Pumping demineralized water at rated pressure and flow to and from the test tank.
- Pumping demineralized water from the test tank into the reactor vessel against existing reactor pressure (this test may only be performed during reactor shutdown) to ensure a functioning flow path.

A tank zero level shutoff prevents damage to pump from insufficient suction. The storage tank, test tank, pump and interconnecting piping and valves are in the RB on the -8.5 m Level in Room 1551 that is nearest to the ICS connection line. There is a single containment penetration for the BIS injection line. The injection line connects with the ICS "C" return line downstream of the two condensate return valves providing a flow path into the reactor vessel.

6.8.2.4 Materials

The following ASME documents contain requirements that are applicable to the BIS:

- ASME B31.1, “Power Piping,” (Reference 6-40), contains piping requirements for the BIS except for the SC1 portion of the system.
- ASME BPVC, “Section III, Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NB - Class 1 Components,” (Reference 6-10), contains design requirements for valves and piping for the SC1 portion of the BIS.
- ASME BPVC, “Section XI, Division 1 – Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” (Reference 6-41), contains requirements for in-service inspection of BIS SC1 components.

The system piping from and including the air-operated injection valve (AOV) to the connection with the ICS is ASME BPVC Section III, Subsection NB. Containment isolation is provided by a single check valve inside containment for immediate isolation and the check valve and AOV for long-term containment isolation. All system piping is either 304 or 316 stainless steel.

All equipment not required for injection, such as system drains, test equipment, and service air and demineralized water are classified as Quality Group D and designed to ASME B31.1 standard and are 304 or 316 stainless steel.

The BIS meets Seismic Class 1-A for the pump suction piping, pump discharge piping and injection piping up to the connection with the ICS. The AOVs and pump are Seismic Category 1-B. The test tank and test loop piping, demineralized water piping, service air piping, and system drains after the isolation valves are non-seismic.

The BIS is classified as a Quality Group D system. Quality Group D system atmospheric storage tanks are designed to API-650, “Welded Tanks for Oil Storage,” (Reference 6-42). The BIS storage tank is constructed from type 304 or 316 stainless steel and contains a hatch, heater system, sparger, and connections for an outlet, overflow, vent, service air and demineralized water inlet, and associated instrumentation.

The injection pump is designed to API-674, "Positive Displacement Pumps – Reciprocating," (Reference 6-43). The base material of the injection pump metal parts in contact with the solution is 300 series stainless steel.

All equipment in the BIS in contact with neutron absorbing solution is stainless steel for corrosion protection.

6.8.2.5 Interfaces with Other Equipment and Systems

System interfaces for the BIS are shown in Table 6-5 and Figure 6-17.

6.8.2.6 System and Equipment Operation

The BIS has four operating modes:

- Standby
- System Injection
- Pump Test
- System Injection Test Mode

BIS functions are required during power operation or reactor startup.

6.8.2.7 Instrumentation and Control

The BIS has the following instrument and control features:

1. High and low storage tank solution temperature annunciates in the control room at temperatures outside the allowed range.
2. High and low storage tank solution level is annunciated in the control room when the level is outside its normal allowed limit.
3. Low storage tank level automatically shuts off the injection pump when the solution level in the storage tank is below the zero level.

Either the storage tank or test tank discharge valve must be fully open for the pump to run.

The following describes key instrumentation features of the BIS:

1. Storage tank solution is measured by a level transmitter. MCR level indication and high/low level alarms provided in the MCR HSI display.
2. Pump discharge pressure is sensed by a single pressure transmitter that provides MCR pressure indication through the HSI display.
3. Storage tank solution temperature is sensed by a temperature element providing MCR storage tank solution temperature through the HSI display.
4. Pump flow is sensed by a single flow element and flow transmitter with flow indication in the MCR.
5. Pump status indication (energized status) provided in the MCR HSI display.
6. System status indication and annunciators through the MCR HSI display for system bypass or inoperable condition.
7. Valve position indication through the MCR HSI display for the storage tank outlet valve and test tank outlet valve.
8. Pump overload trip or power loss indication provided through the MCR HSI display.

A single pressure gauge is locally mounted for pressure indication during the pump test mode and is visible from the test loop control valve aiding control valve positioning.

<u>Main Control Room Operation</u>

Manual initiation, operation, and control of the BIS is from the MCR through the HSI display. The means for manual actuation and for monitoring shutdown status from the MCR meets SSR-2/1 (Reference 6-13), addressing monitoring and operator action of shutdown systems.

BIS actuation is dual-action initiation through the HSI display in the MCR that guards against accidental or inadvertent actuation of the system ensuring intentional system actuation.

Controls for the BIS are located through the HSI display in the MCR facilitating system operation. System and component operating status, including any system bypasses, system out of service, or manual overrides is provided. Manual system initiation and shutdown is provided.

The following BIS displays, alarms and controls are provided in the MCR HSI display:

System Annunciators

- High and low alarms for storage tank solution temperature
- High and low alarms for storage tank solution level
- Manual or automatic system out-of-service condition

System Indication

- Storage tank solution level
- Pump discharge pressure
- Pump operation (on/off)
- Storage tank and test tank outlet valves position (open/closed)
- Storage tank solution temperature
- System manually out of service
- Pump overload trip or power loss
- Pump flow rate

<u>Local Panel Operation</u>

A local panel is provided for the following:

Controls:

- Heater controls

Indications:

- Status indications for the mixing heater
- Storage tank solution temperature
- Storage tank solution level

<u>Secondary Control Room</u>

The BIS is initiated from the SCR if the MCR is not operational.

6.8.2.8 Monitoring, Inspection, Testing and Maintenance

The BIS is functionally tested to ensure the pump develops rated flow and discharge pressure and the flow path from the pump suction to the RPV without contaminating the reactor with neutron absorber solution during each planned outage or reactor shutdown for refuelling. Functional testing is performed by circulating demineralized water from the test tank, return to the test tank for the flow and pressure test, and with the injection valve open for the RPV injection test.

Maintenance Provisions

The BIS design is provided with adequate equipment removal paths and personnel access for repair and replacement. The following features are provided to facilitate component maintenance:

1. The relief valve is provided with flanged inlet and outlet connections facilitating removal for bench testing.
2. Isolation valves are provided on the pump suction and discharge side for maintenance.
3. The storage tank heater can be removed without draining the tank.
4. Sufficient pull space is provided for removing/replacing the storage tank heater.
5. The storage tank is fitted with a top-mounted entry hatch and an external ladder for equipment installation, maintenance, and chemical addition.
6. In the event of a sodium pentaborate system injection or the sodium pentaborate leakage into the system piping, the demineralized water supply can be used to flush the piping downstream the storage tank outlet valve.
7. Sufficient pump room is provided for removing plungers, pistons, rods, crankshaft, and inspecting parts.
8. Ample head room is provided above the pump for a crane, hoist, or tackle.
9. Sufficient space is provided around the storage tank for inspection and maintenance.
10. Sufficient space is provided for access and local instrument calibration.
11. System maintenance is performed by closing the storage tank outlet isolation valve.
12. A maintenance valve downstream of the inboard containment isolation valve can be closed allowing system maintenance and isolation valve testing.
13. Test vents and drains for testing CIVs.

Dedicated piping and collection for sodium pentaborate solution is provided where drainage occurs. The collection vessel is a portable stainless-steel drum. A low containment surrounds the system preventing the spread of sodium pentaborate leaks. No special temporary or permanent provisions such as ladders, scaffolding, overhead cranes, etc., are required for maintenance.

Surveillance Testing and In-Service Inspection Provisions

All BIS components can be tested during normal plant operation. Pre-operational tests are conducted demonstrating the system flow path, adequate pump Net Positive Suction Head (NPSH), and pressure drops in accordance with system design documents.

Pre-operational and periodic hydrostatic testing of the system per OLC are performed complying with ASME BPVC Section III. Periodic testing of the system and components is scheduled at a frequency commensurate with the OLC and meets the requirements of SSR-2/1 (Reference 6-13).

6.8.2.9 Radiological Aspects

There are no radiological releases from a failure in the boron injection system. Breaks outside containment for the injection flow path through the injection location in the condensate return line of the ICS "C" loop downstream of the two ICS condensate return valves through a separate containment penetration is evaluated in PSR Ch. 15, Section 15.5.9.2.3.

6.8.2.10 Performance and Safety Evaluation

The BIS injects a neutron absorber quantity into the reactor vessel producing a minimum concentration to achieve a cold shutdown accounting for leakage and imperfect mixing.

The required shutdown concentration is achieved in a mass of water equal to the sum of the mass of water in the reactor vessel plus the mass of water in the SDC system.

The boron injection rate is based on a boron concentration rate of change between 8 to 20 ppm/minute in the reactor water that includes the weight of water in the reactor and shutdown cooling loops at normal level and at 20 °C (68 °F). The BIS pump can inject the sodium pentaborate solution at all reactor pressures from 12.41 MPaG (1800 psig) (vessel bottom) to zero MPaG.

The BIS can be actuated and operated in the event of LOOP. BIS contains safety features such as containment isolation and reactor coolant pressure boundary piping (Class 1 piping) for the injection line. The BIS injection line is part of the ICS described within Section 6.2.1.

6.8.3 Ultimate Pressure Regulation

Design Extension Condition (DEC) complex sequences that are associated with RCPB overpressure may be postulated during final design activities. Additional design features that are necessary to address such sequences could include the Ultimate Pressure Regulation subsystem of the ICS. PSR Ch. 15.9, Appendix 15D discusses risk reduction features for DEC complex sequences.

In the event of some Design Extension Complex Sequences such as a full or partial Anticipated Transient Without Scram, the UPR subsystem of the ICS functions to lower both RPV pressure and level which lowers core power and instability. It is a DL4b function and the UPR is classified as SC3. During this event, the RPV is isolated, and the ICS trains operate as designed.

The UPR subsystem includes components in each train's steam supply line. It is designed with positive shutoff means during normal operations or design basis events to ensure zero leakage. It initiates automatically on a high RPV pressure that is above the ICS initiation setpoints for design basis events. The UPR operates in tandem with the control rod drive motor run-in feature and the BIS to achieve a long-term reactor shutdown state and is considered a backup to the hydraulic Scram function.

6.8.4 Recombiners

When an ICS train is placed in service, radiolytic gases are removed from the ICS train by passive autocatalytic recombination. Figure 6-17 illustrates a preliminary configuration with a recombiner positioned on an IC.

Table 6-1: ICS Structure, System, and Component Classification

Principal Components	Safety Class	Location	Quality Group	Seismic Category
Steam supply, condensate return, standby gas purge piping	SC1	CV	A	A
SDC interface piping to containment isolation valve, A and B trains	SC1	CV, RB	A	A
BIS interface piping to BIS interface valve, C train	SC1	CV	A	A
SDC interface piping from containment isolation valve to downstream redundant isolation valve, A and B trains	SC1	CV	B	A
ICS pools atmospheric vent piping	SC1	RB	B	A
Outer pool to inner pool cross-connect piping	SC1	RB	B	A
Long term ICS pool makeup piping (also referred to as flex-makeup piping)	SC3	RB	D	A
Flow detection differential pressure instrumentation impulse piping	SC1	CV, RB	B	A
Isolation Condensers (1)	SC1	CV, RB	A, B	A
Condensate return valves (Open/Closed Only)	SC1	CV	A	B
Condensate return valves (Throttling)	SC2	CV	A	B
Standby gas purge valves	SC1	CV	A	B
Containment isolation valve to SDC system, A and B trains	SC1	CV	A	B
Outer pool to inner pool cross-connect backflow prevention devices	SC1	RB	B	B
Redundant isolation valve to SDC system, A and B trains	SC1	CV	B	B
Flow detection differential pressure instrumentation, including tubing, cables, trays, and instrument racks	SC1	RB, CV	—	A and B
Wide range pool level instrumentation used for post-accident monitoring, long term (>72 hours)	SC1	RB	—	B
All piping installed temperature instrumentation, pool temperature instrumentation and narrow range pool level instrumentation used	SC3	RB, CV	—	Non-Nuclear Seismic

Principal Components	Safety Class	Location	Quality Group	Seismic Category
Pneumatic supply tubing and components from the actuator to the control solenoid valves for the open/closed only condensate return valves, containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, CV	—	A
Hydraulic supply tubing and components from the actuator to the control solenoids valves for the throttling condensate return valves	SC2	CV	—	A
Control solenoid valves for the condensate return valves, containment isolation valves, and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, CV	—	B
Control solenoid valves for the throttling condensate return valves	SC2	CV	—	B
Pneumatic supply tubing and components from the interface point with Plant Pneumatics System to the control solenoids valves for the condensate return valves	SCN	RB, CV	—	Non-Nuclear Seismic
Pneumatic supply tubing and components from the interface point with Plant Pneumatics System (not including the check valve) to the control solenoids valves for the containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC3	RB	—	A
Hydraulic supply tubing from the positioner to the control solenoid valves for the throttling condensate return valves	SC3	CV	—	A
Pneumatic supply check valve to the valve actuators for the containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC3	RB	—	B

Notes:

(1) The steam supply components from the steam supply piping to and including the flow restrictors, and the condensate return components from and including the 6” x 8” tee to the condensate return piping are Quality Group A. All components between the steam flow restrictors and the condensate tee including headers, drums, tubes, flanges, and covers are Quality Group B.

Table 6-2: ICS System Interfaces

Interfacing System	Interface Description	Interface Boundary
Nuclear Boiler System	Decay heat removal, receive steam, return condensate. Accept standby mode IC gas purge.	RIVs, steam supply and condensate return RPV nozzles. Gas purge to main steam lines at a location between the RIVs and the containment penetration.
Safety Class 1 Instrumentation and Control	Safety Class 1 Instrumentation and Control System to receive/provide signals to Safety Class 1 instrumentation and controls in ICS.	Interface boundary is at the input/output terminations in the Distributed Control and Instrumentation System cabinets within the scope of the SC1 C&I control equipment.
Safety Class 2 and Safety Class 3 Instrumentation and Control	Safety Class 2 and Safety Class 3 Instrumentation and Control System to receive/provide signals to Safety Class 2 and Safety Class 3 instrumentation and controls in ICS.	Interface boundary is at the remote multiplexer unit associated with the instrument or control equipment.
Process Radiation and Environmental Monitoring System	Monitor ICS pool compartments for elevated radiation levels that may indicate ICS pressure boundary leakage.	Radiation detectors mounted in the ICS pool compartments or atmospheric vent line.
Boron Injection System	ICS serves as a boron injection point via the train C condensate return line to the RPV chimney interior. The ICS condensate return piping for train C provides the interface between the BIS and the RPV. This interface avoids the need dedicated RPV penetration for BIS. The ICS provides the interface piping that extends from the condensate return line to a system boundary isolation valve that is in the scope of the BIS and is located inside the containment vessel. The ICS interface piping to the BIS connects to the condensate return piping between the condensate return valves and the RIVs at the lowest horizontal section of piping that includes the return isolation valves, effectively in the zone of the loop seal. This ensures that all of the interface piping to the BIS contains subcooled water.	Pipe connection on the ICS train C condensate return line at a location between the condensate return valves and the RIVs.

Interfacing System	Interface Description	Interface Boundary
ICS Pool Cooling and Clean up System	Provide ICS pool water continuous cleanup to meet water quality standards and maintains the pool water below the temperature limit. Also, provides normal operational makeup water to maintain pool water levels.	The interface occurs through seismically qualified suction and discharge connections to the ICS pools.
Shutdown Cooling System	ICS serves as a Shutdown Cooling suction point via the trains A and B condensate return lines from the RPV chimney interior. The ICS condensate return piping for trains A and B provides the interface between the SDC system and the RPV. This interface avoids the need for two RPV penetrations dedicated to shutdown cooling. The ICS provides the interface piping that extends from the condensate return line and out through the containment vessel where the connections to SDC occur. There are two ICS scope remote-actuated isolation valves outside of containment in each interface line. (Requirements pertaining to the outside containment isolation valves are located in PSR Ch. 5, Section 5.1.6.) The SDC system begins at the outlet of the most outboard valve. This arrangement maintains the SC1 and SC2 piping and components within the scope of the ICS. The ICS interface piping to the SDC system connects the condensate return piping between the condensate return valves and the RIVs in the lowest horizontal section of piping that includes the return isolation valves, effectively in the zone of the loop seal. This ensures that all of the interface piping to SDC contains subcooled water.	Pipe connection on the ICS trains A and B condensate return lines at a location between the condensate return valves and the RIVs.
Liquid Waste Management System	Provide a storage vessel for ICS pool water that may need to be relocated to facilitate component maintenance or inspection that is located within the pool compartments, such as the ICs.	At this time the interface is made by temporarily installed pump taking suction from the refueling floor elevation and discharging through a flexible connection to installed piping that is in the scope of LWM system.

Interfacing System	Interface Description	Interface Boundary
Plant Pneumatics System	Provide pressurized nitrogen gas to charge accumulators for the condensate return valve actuators and the SDC interface isolation valve actuators. The function of supplying nitrogen to the condensate return valves is non-safety class. The valves are held closed for the standby mode by nitrogen pressure, so the plant pneumatics function is important to prevent inadvertent actuation of the system. An accumulator and its inlet check valve, which are in the scope of ICS, store an adequate supply of nitrogen as a coping measure in the event that the plant pneumatics nitrogen source is lost. Since the condensate return valves fail to the open or safe position on loss of pneumatics, the nitrogen supply piping, accumulator, and check valve are not SC components	Interface boundary is at the inlet of the Safety Class N backflow preventors.
Safety Class 1 Electrical Distribution System	Safety Class 1 Electrical Distribution System and batteries provide power to Safety Class 1 electrical loads in the ICS system.	Interface boundary is at the input to the local breaker at the Motor Control Center termination.
Primary Containment System	Provides the piping penetration sleeves and connections to ICS system piping ensuring a leak tight barrier.	Interface occurs at the penetration sleeve to piping weld.
Reactor Building Structure (RBS)	Provides ICS pool compartments and ICS support structure overall support (e.g., IC supports, piping supports)	Interface occurs at the ICS component support to RB structure attachment points (e.g., fasteners, welds).

Table 6-3: Preliminary Primary Containment Key Design Parameters

Parameter	Value and Units
Free Containment Volume (estimated)	{7311.3 m^3 (258,196 ft^3)}
Minimum Free Volume (design)	{6960 m^3 (245,790 ft^3)}
Containment Height, excluding the dome	{37.985 m (124.6 ft)}
Containment Diameter (outer)	{19.28 m (63.3 ft)}
Containment Diameter (inner)	{17.45 m (57.3 ft)
Containment Shell Plate Thickness	{15 mm (0.59 in)}
Containment Wall Thickness	{915 mm (36 in)}
Containment Closure Head Diameter	{6.8 m (22.3 ft)}
Number of Manholes on Bulkhead	{2 (Assumed)}
Diameter of the Manholes	{0.762 m (2.5 ft) (Assumed)}
Containment Atmosphere	Nitrogen with less than 4% Oxygen
Normal Pressure	{1.72 - 9.0 kPaG (0.25 - 1.30 psig)}
Containment Design Internal Pressure (1)	413.7 kPaG (60.0 psig)
Containment Maximum Negative Pressure (2)	{-13.8 kPaG (-2 psig)}
Max. Bulk Average Temperature (3) (Normal Operating Temperature)	{57.2 ºC (135 ºF)}
Max. Average Temperature – Upper Containment	{66 ºC (150 ºF)}
Max. Average Temperature – Lower Containment	{43 ºC (110 ºF)}
Peak Transient Shell Temperature: 0.0 – 0.01 h	{195 ºC (383 ºF)}
Peak Transient Shell Temperature: 0.01 - 72 h (4) (Design Temperature)	165.5 ºC (330 ºF)
Containment Design Leakage Rate	0.35 wt.%/day

Notes:

(1) Design pressure is conservatively established with approximately 10-15% margin above the initial thermal-hydraulic analysis peak pressure and the historical BWR Mark I and II containment design pressure values.

(2) Containment maximum negative pressure does not apply to DEC.

(3) Bulk temperature is representative of normal plant operating condition based on historical BWR steel containment temperature records. Bulk average temperature is not indicative of local component thermal environment maximum operating temperature, which is separately evaluated.

(4) Peak transient temperature is also used for environmental qualification. Peak temperature for the most severe conditions of beyond the design basis events may exceed this value for a short duration.

Table 6-4: Primary Containment System Interfaces

Interfacing System	Interface Description	Interface Boundary
Multiple MPLs	Process systems penetrating containment provide containment isolation. Reference Penetration Interfaces in Figure 6-9 : Primary Containment Simplified Diagram	For mechanical penetrations, the interface boundary is at the weld between the penetration assembly and containment structural steel. See table end notes for clarification on suggested approach. (1, 2, 3)
Multiple MPLs	PCS to provide structural support of process piping systems within containment.	Attachment point of support or anchor.
Nuclear Boiler System (B21)	NBS to provide attachment / support of refueling bellows seal.	Support bracket / ring on RPV exterior beneath RPV upper head flange elevation.
Nuclear Boiler System (B21)	PCS to provide vertical and lateral RPV support.	Support interface brackets / lugs on RPV exterior.
Safety Class 1 Instrumentation and Control System (C10)	SC1 Instrumentation and Control System to receive / provide signals to SC1 instrumentation and controls in T10 system.	Interface boundary is at the Remote Multiplexer Unit (RMU) associated with the instrument or control equipment. (4)
Non-Safety Class (SCN) Instrumentation and Control System (C40 to C45)	SCN Instrumentation and Control System to receive / provide signals to SCN instrumentation and controls in T10 system.	Interface boundary is at the RMU associated with the instrument or control equipment. (4)
Process Radiation and Environmental Monitoring System (PREMS) (D11)	PREMS to monitor containment atmosphere for oxygen, hydrogen, temperature, pressure, and water level.	Mechanical penetration.
Process Radiation and Environmental Monitoring System (D11)	PREMS to provide fission product and area radiation monitoring within containment.	Mechanical penetration.
Isolation Condenser System (E52)	Ultimate Pressure Regulation (UPR) discharge of RPV inventory to T10.	UPR inside containment.
Fuel Pool Cooling and Cleanup System (G41)	Fuel Pool Cooling and Cleanup System to provide supply water into PCCS.	Supply connection in equipment pool. (5)
Fuel Pool Cooling and Cleanup System (G41)	PCCS to return heated water into the Equipment Pool.	Return connection in equipment pool.

Interfacing System	Interface Description	Interface Boundary
Plant Pneumatics System (P52)	PCCS Air-Operated Valves.	PCCS valves outside of containment.
Non-Safety Electrical Distribution System (R30)	SCN Electrical Distribution System to provide power to SCN electrical loads in T10 system.	Interface boundary is at the input to the local breaker at the MCC termination. (4)
Containment Inerting System (T31)	CIS to supply nitrogen or fresh air to containment during inerting and de-inerting evolutions.	CIS supply penetration.
Containment Inerting System (T31)	CIS to exhaust nitrogen or fresh air from containment during inerting and de-inerting evolutions.	CIS exhaust penetration.
Containment Inerting System (T31)	CIS to provide Containment Overpressure Protection.	Exhaust penetration. (6)
Containment Cooling System (T41)	CCS to maintain containment temperature within prescribed band during normal operation.	Containment air coolers.
Cranes, Hoists, and Elevators System (U31)	Monorails and hoists to support transportation of equipment in containment.	Monorails and hoists in containment.
Equipment and Floor Drain System (U50)	PCS to provide structural support of containment sump.	Sump structural support interface with containment.
Equipment and Floor Drain System (U50)	PCS to monitor outflow from containment sump to allow for trending and identification of excessive leakage within containment.	Containment sump belongs to U50.
Equipment and Floor Drain System (U50)	PREMS to monitor containment sump level to allow for identification of excessive leakage within containment.	Containment sump belongs to U50.
Reactor Building Structure (U71)	RB to provide structural support of the SCCV.	Interface boundary at SCCV exterior surface.

Notes:

(1) Individual penetrations are formally owned by and appear on equipment lists for the penetrating process system. The design of penetrations is an integral effort between civil / structural and mechanical systems to address interface with containment structure and process system. Standard penetration drawings will be applied across systems with procurement of penetrations from all systems in a compiled package.

(2) CIV configuration, allowable leakage rate, and leak rate testing configuration are to be incorporated in the system design for each penetration.

(3) Overpressure protection is to be provided for penetrations with trapped liquid volume between isolation valves.

(4) The interface boundary is per 006N4173.

(5) Heated temperature of water maintained below a maximum temperature of 43.3 °C (110 °F) during normal operation.

(6) Relief valve and rupture disc arrangement.

Table 6-5: Boron Injection System Interfaces

Interfacing System	Interface Description	Interface Boundary
Water, Gas, and Chemical Pads	Provide demineralized water for initial storage tank fill and chemical mixing, system flushing, test tank fill for system testing, and maintaining the system full during standby.	The interface is at the upstream side of BIS pressure control valve.
Plant Pneumatics System	Provide service air for periodic storage tank solution mixing through the air sparger in the tank.	The interface is at the upstream side of BIS pressure control valve.
Safety Class 2 and Safety Class 3 Electrical Distribution System	Electrical power for the BIS pump.	The interface boundary is at BIS equipment terminals.
Non-Safety Electrical Distribution System	Electrical power for the storage tank electric heater.	The interface boundary is at BIS equipment terminals.
Electrical Power	Provide electrical power for operation of the storage tank heater, and injection pump.	The interface boundary is at BIS equipment terminals.
Reactor Building Structure	Provide structural support and protection for BIS equipment.	The interface is at the equipment foundation and support structures for the BIS equipment.
Safety Class 2 and Safety Class 3 Instrumentation and Control	Provide for input and output for BIS instrumentation and controlled components.	The interface is at the DCIS.
Isolation Condenser System	Provide an injection path to the reactor for the neutron absorber solution and system isolation from the reactor system.	At the ICS “C” loop return piping.

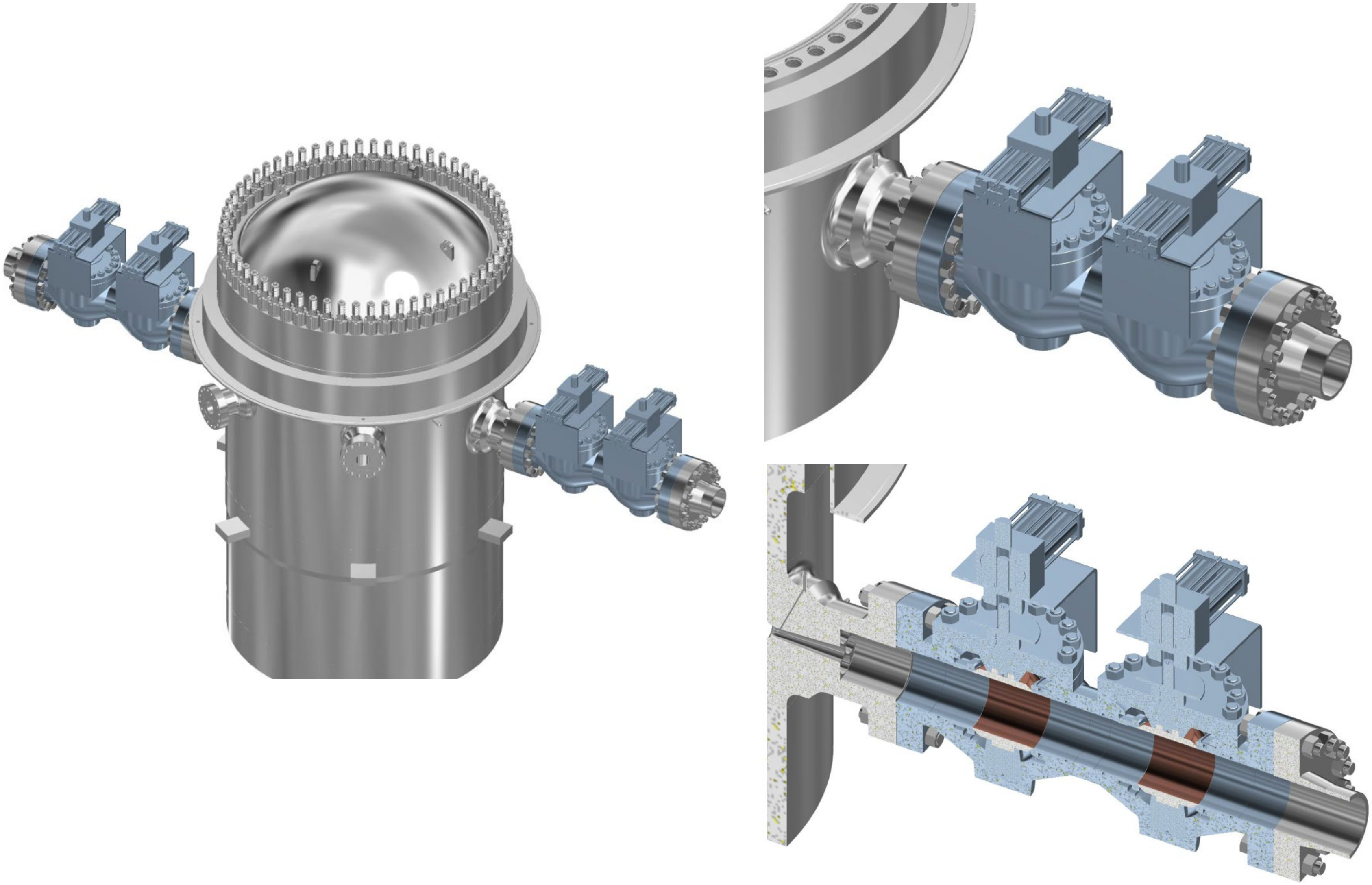

Figure 6-1: Reactor Isolation Valve Assembly Example

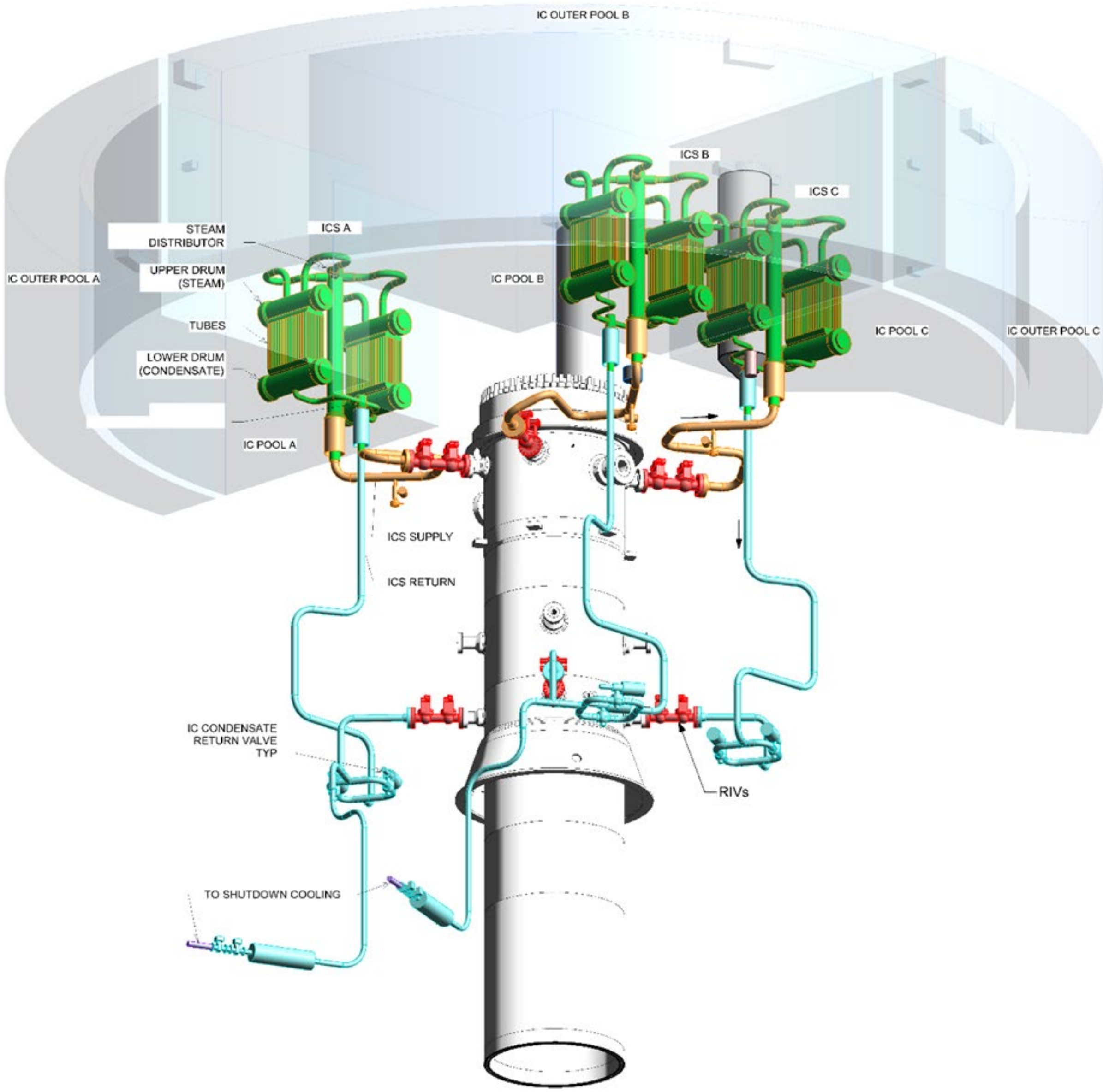

Figure 6-2: Isolation Condenser System Configuration

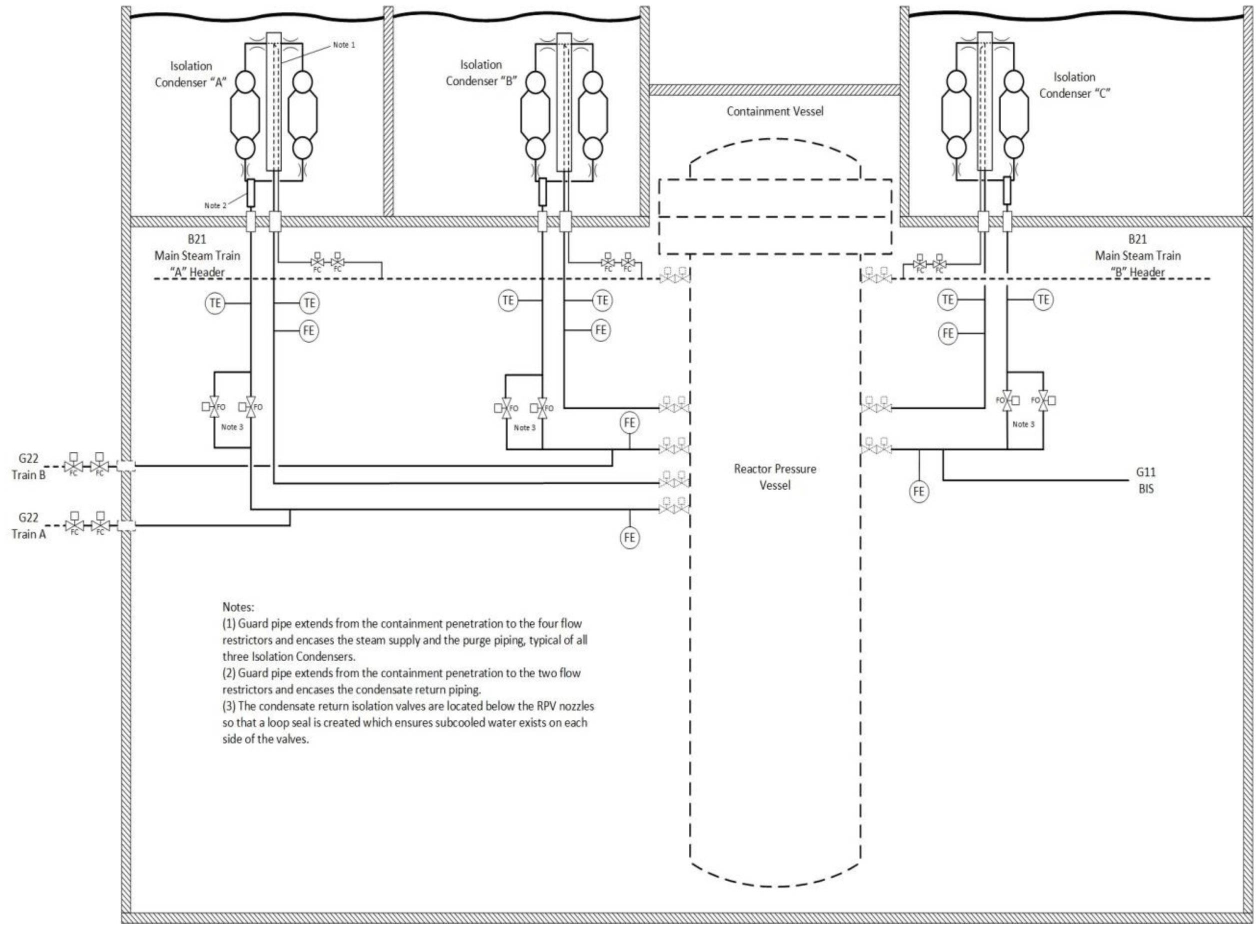

Figure 6-3: Isolation Condenser System Simplified Diagram

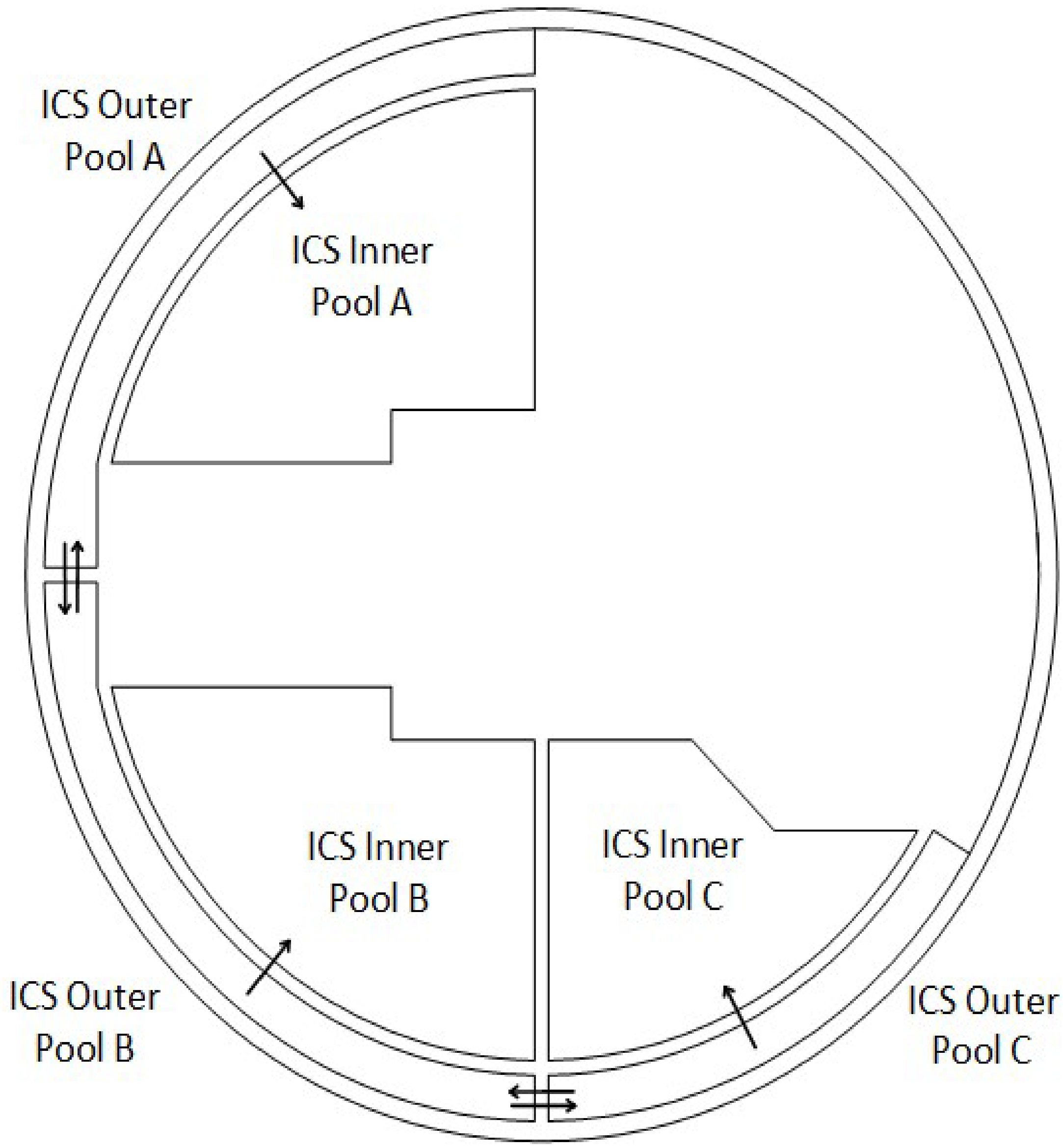

Figure 6-4: ICS Pools Simplified Flow Diagram (Pools)

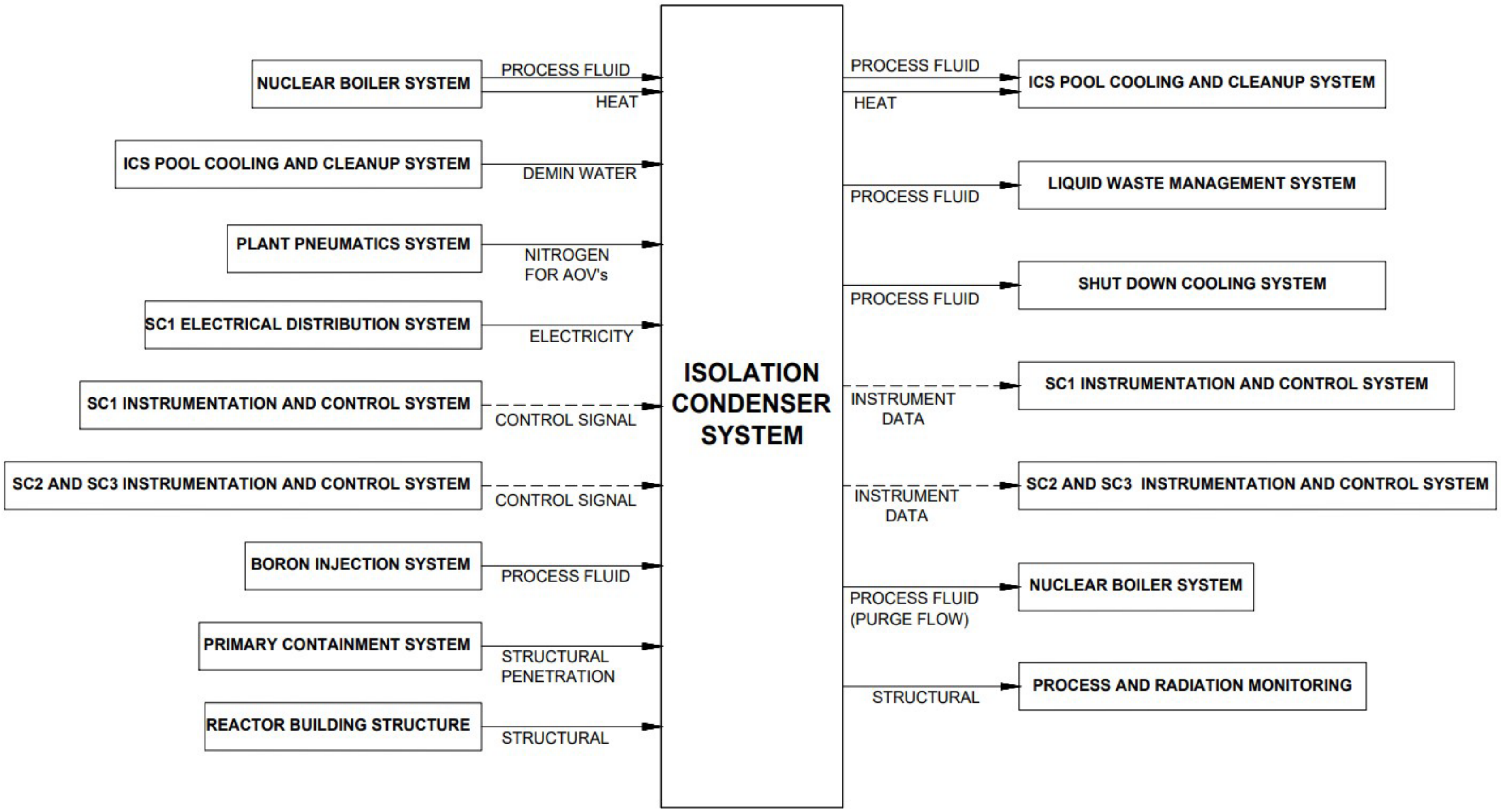

Figure 6-5: ICS System Interfaces

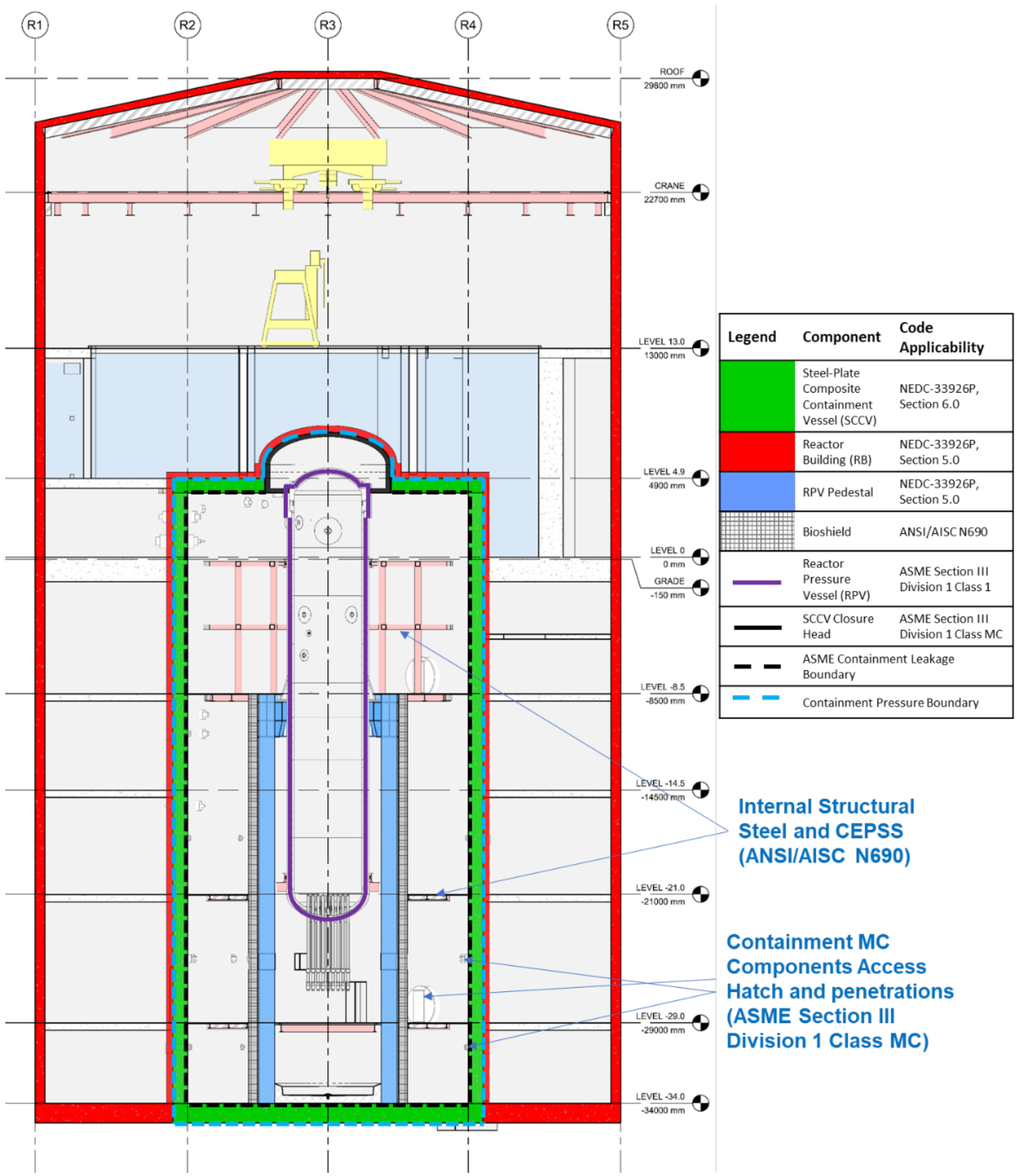

Figure 6-6: General Containment Arrangement – Reactor Building Section View

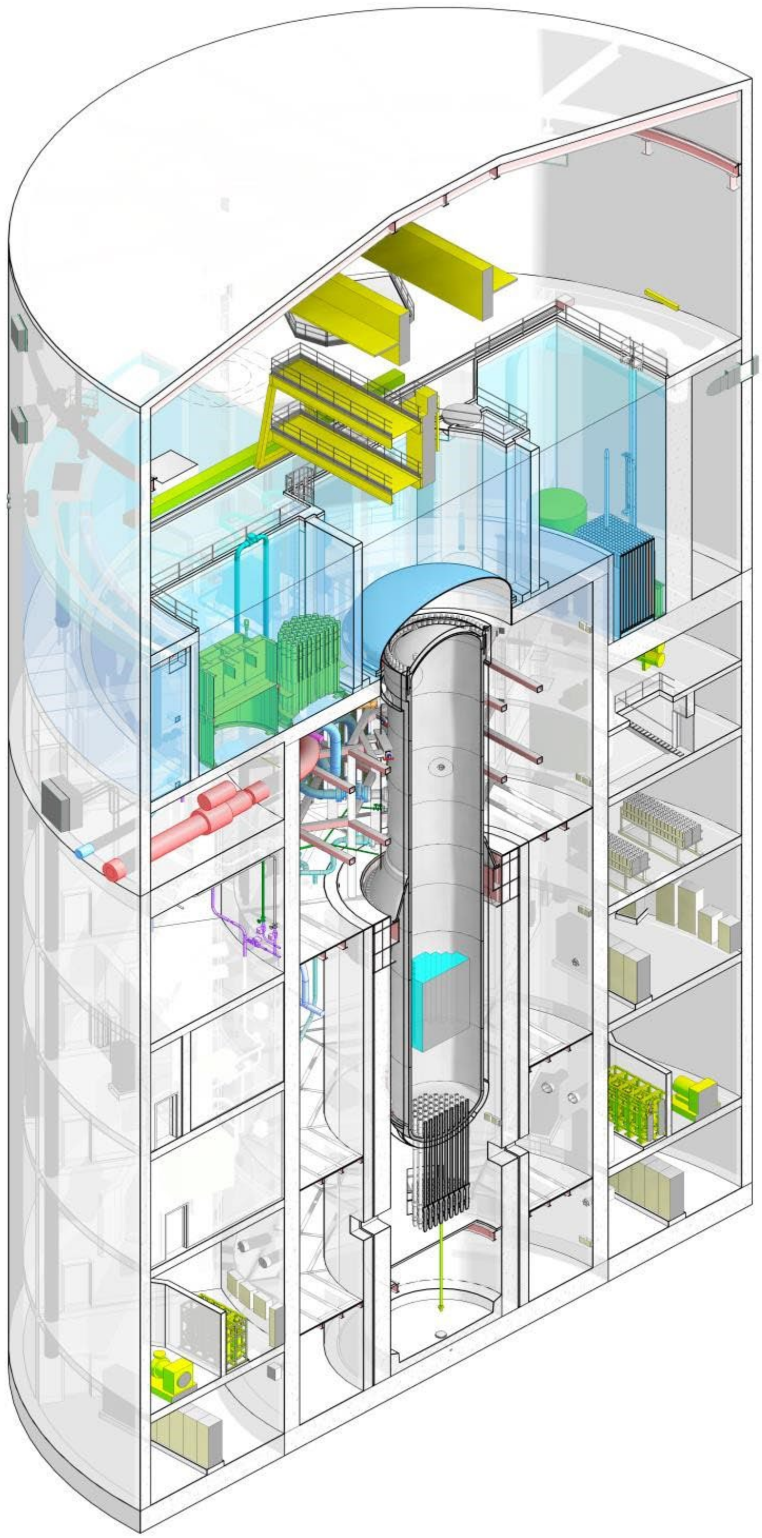

Figure 6-7: General Arrangement – Reactor Building Section View

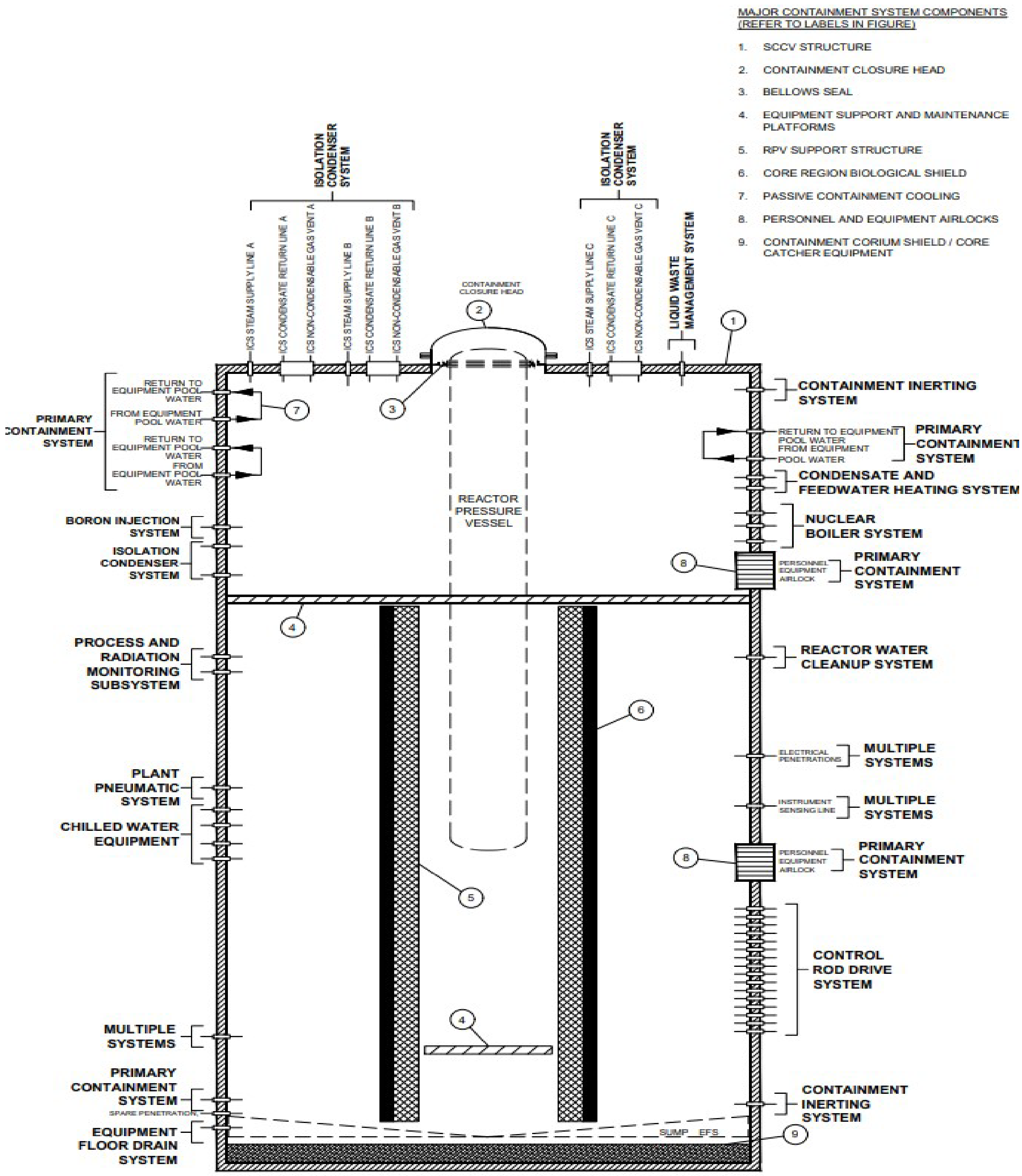

Figure 6-8: Primary Containment System Simplified Diagram (Containment Mechanical Penetrations)

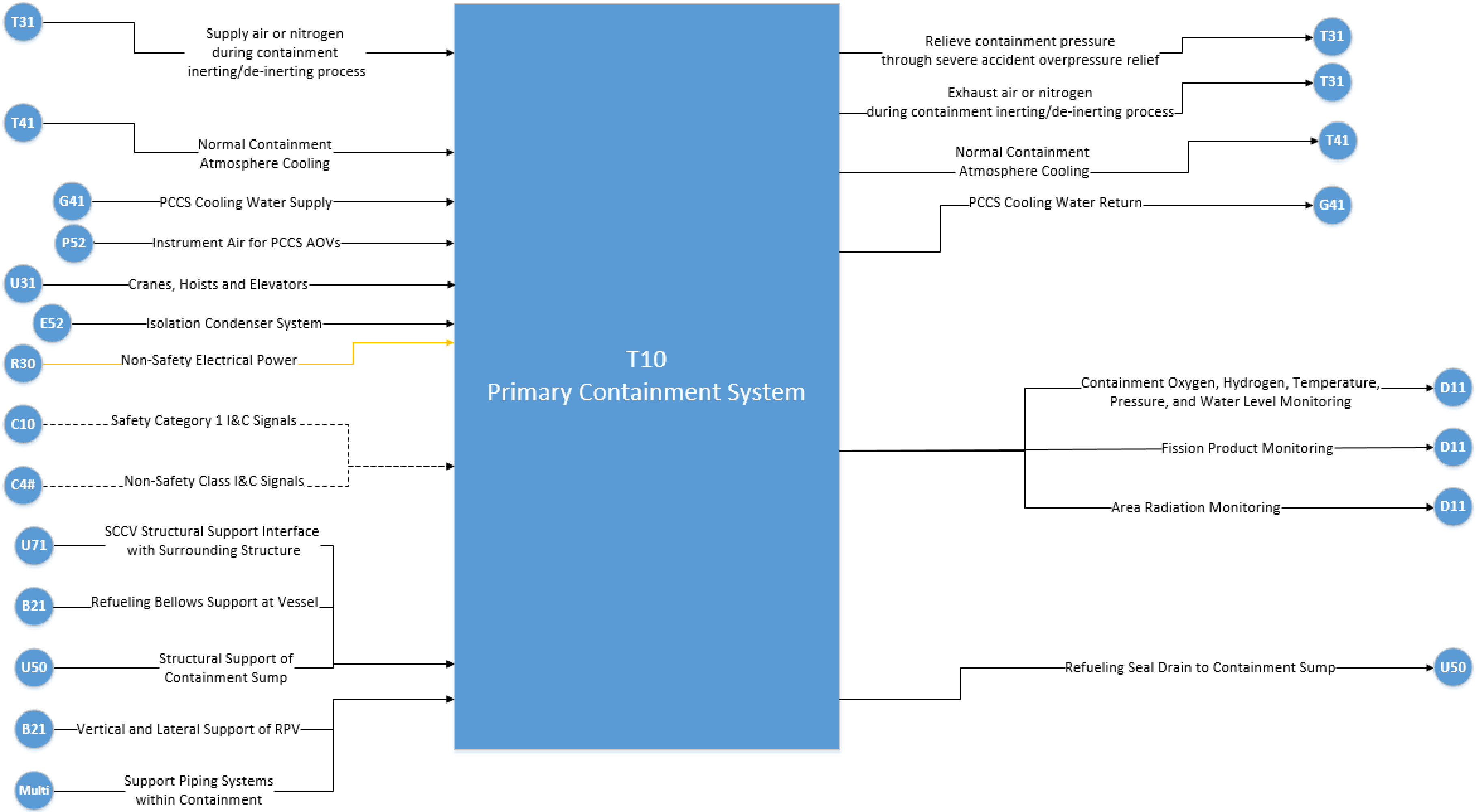

Figure 6-9: Primary Containment System Interfaces

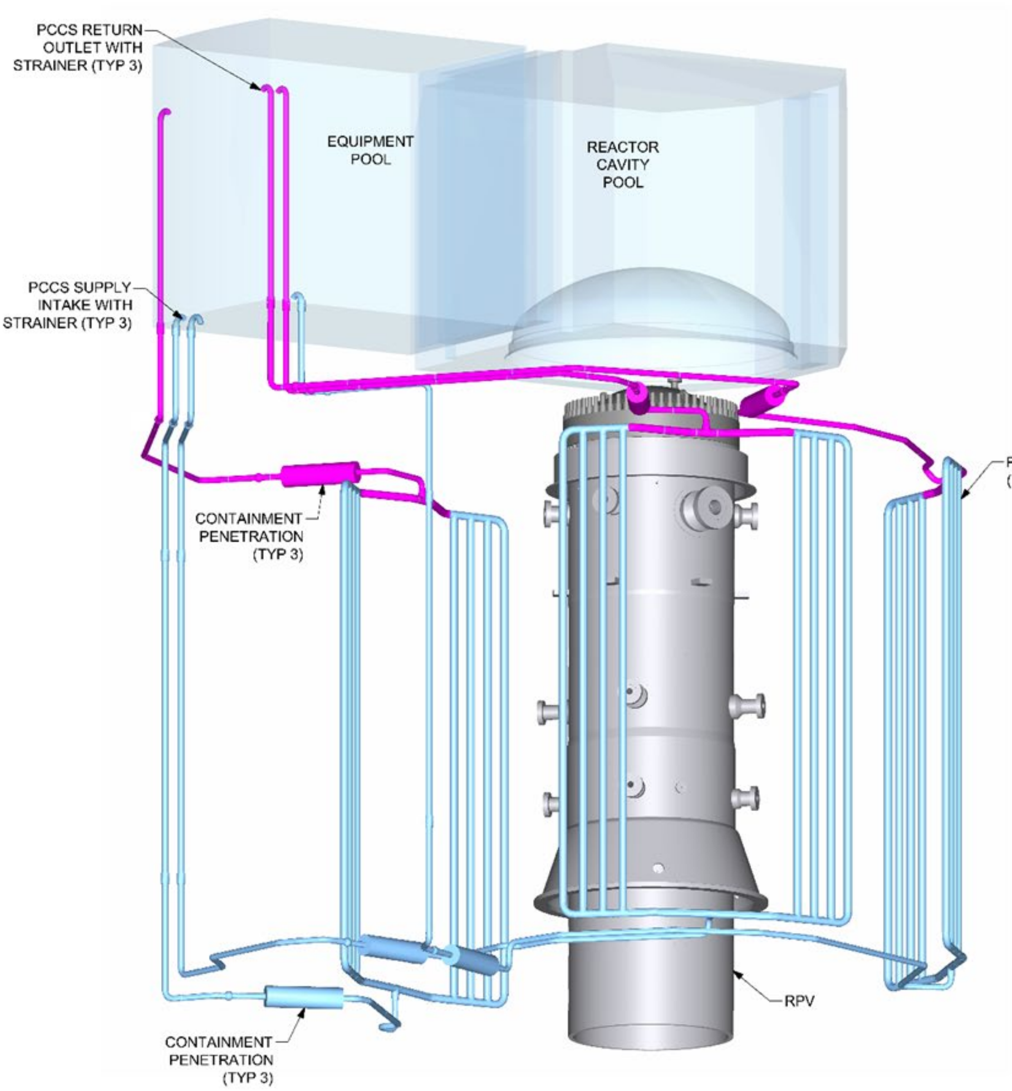

Figure 6-10: Passive Containment Cooling System

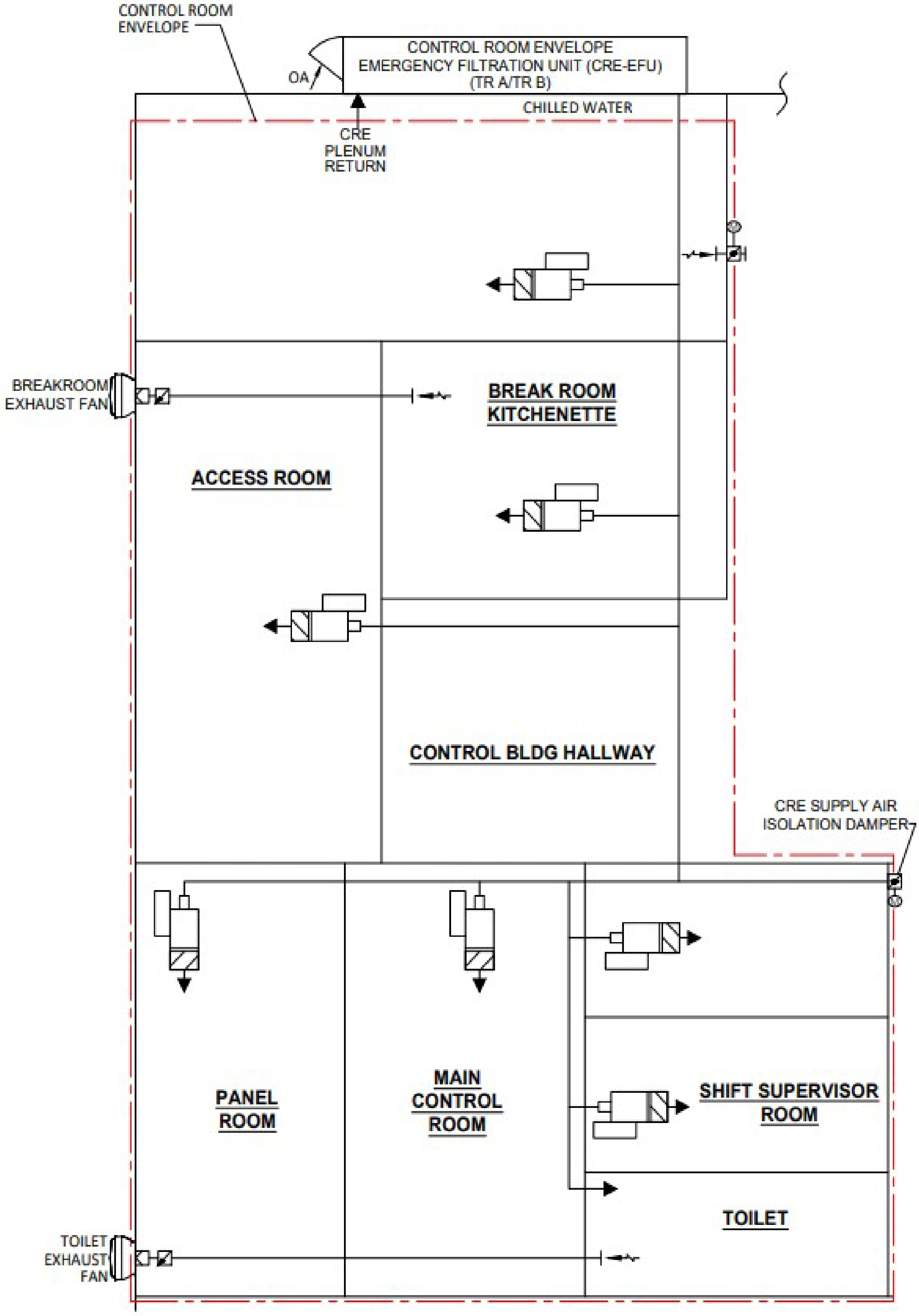

Figure 6-11: Main Control Room Habitability Envelope

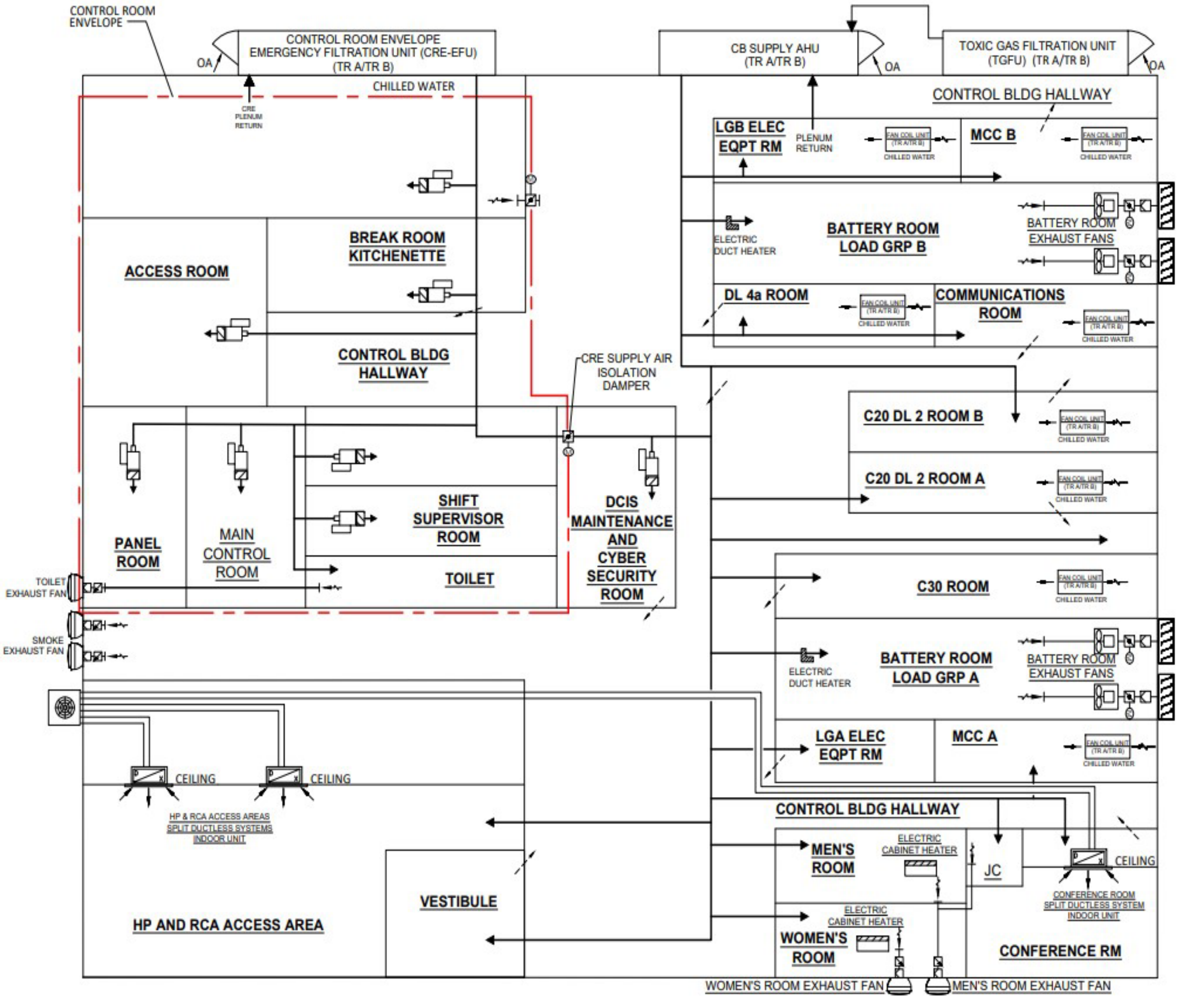

Figure 6-12: Control Building HVAC Simplified Diagram

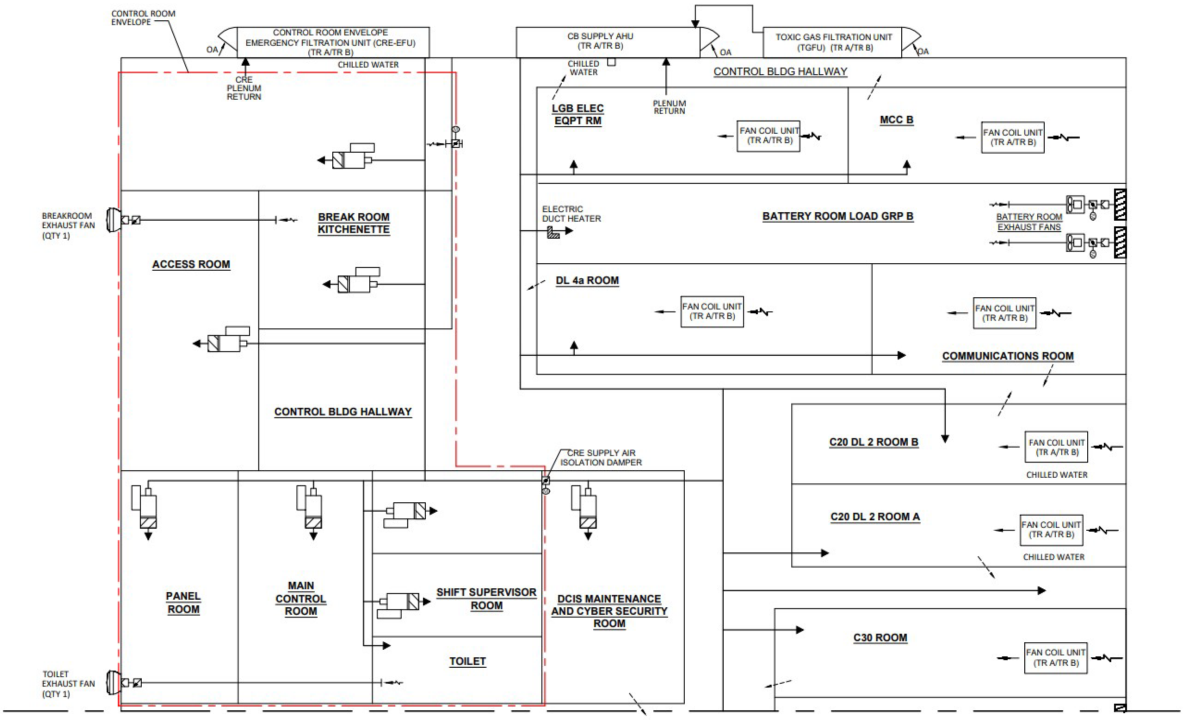

Figure 6-13: Control Building HVAC Simplified Diagram

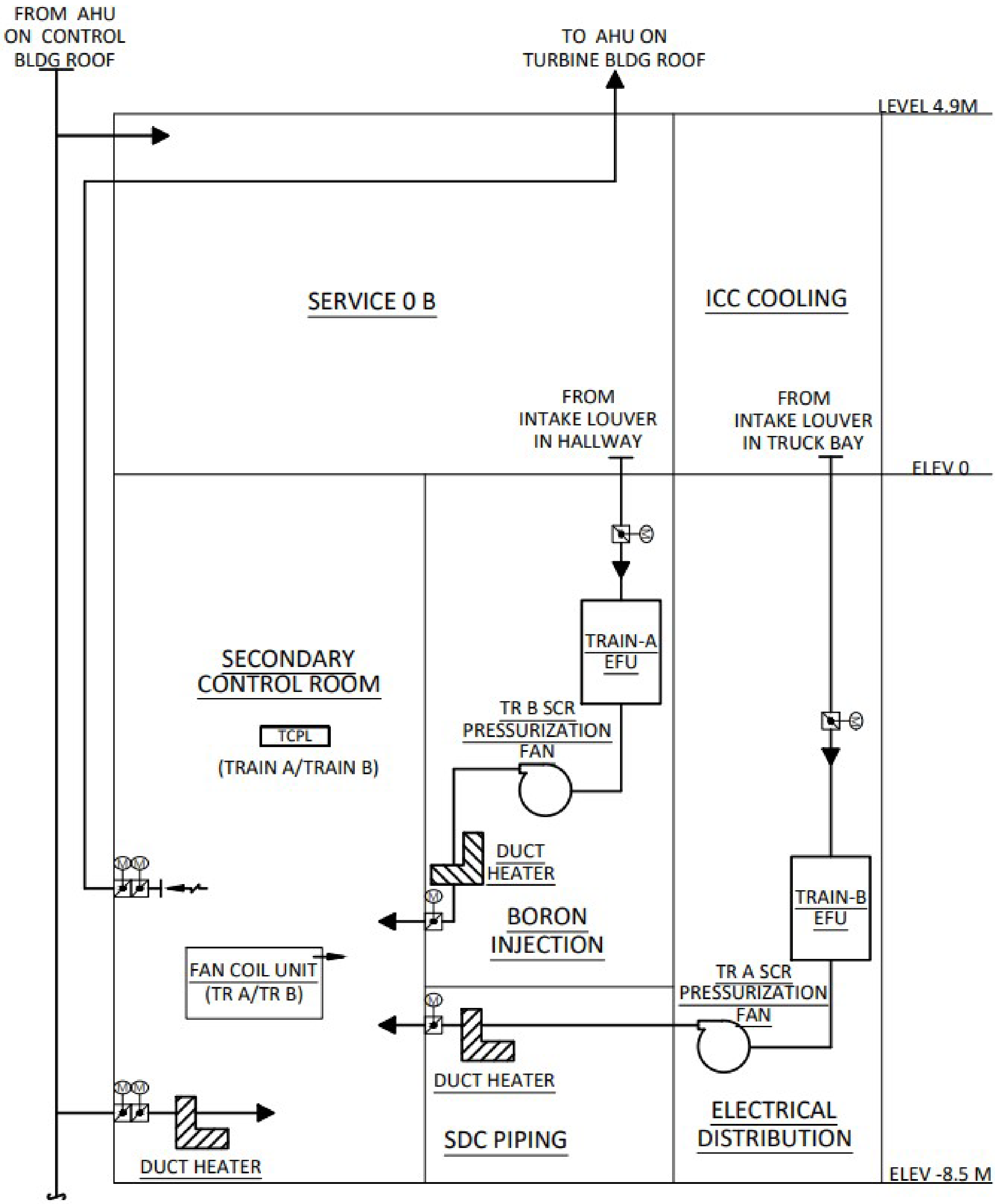

Figure 6-14: Reactor Building Secondary Control Room

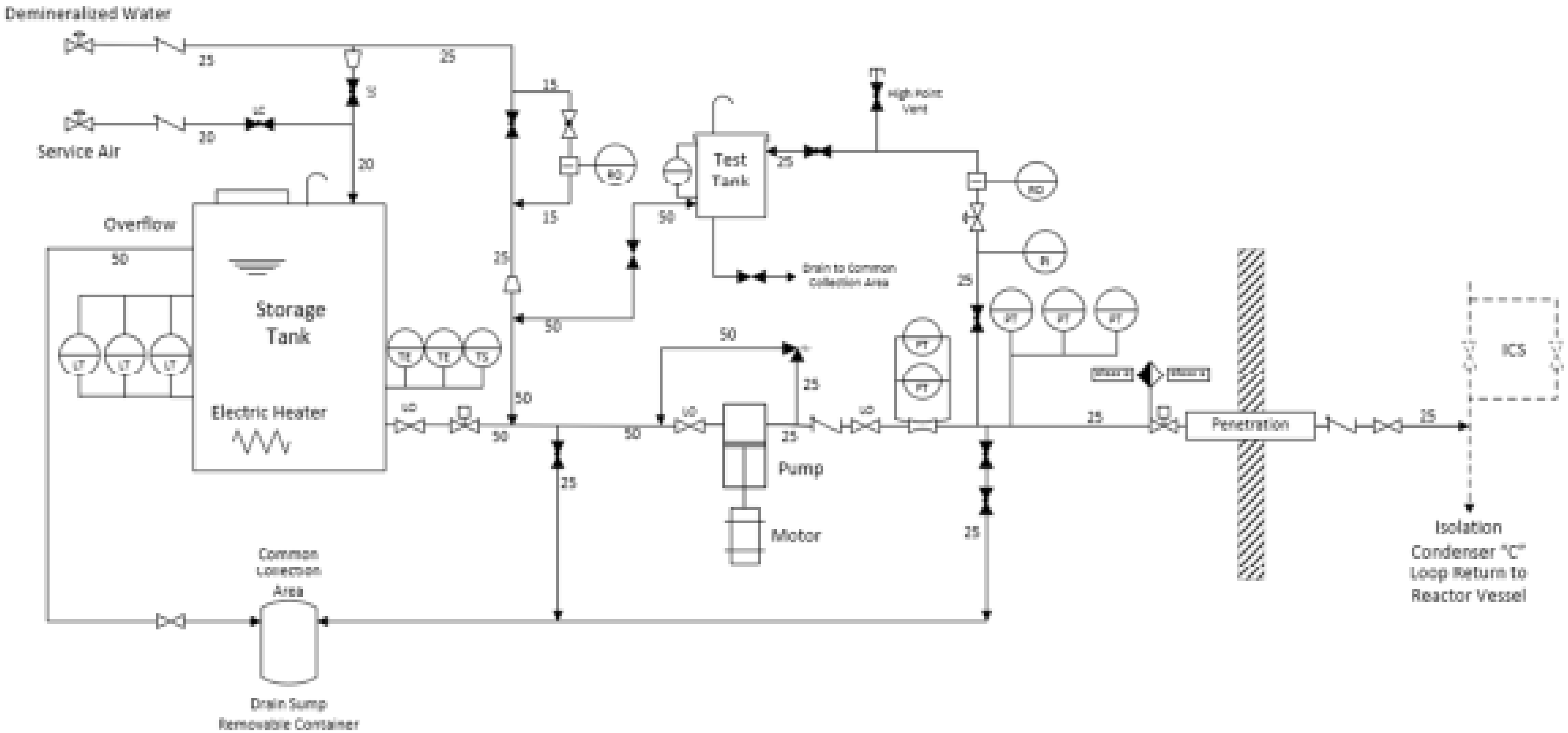

Figure 6-15: Boron Injection System Simplified Flow Diagram

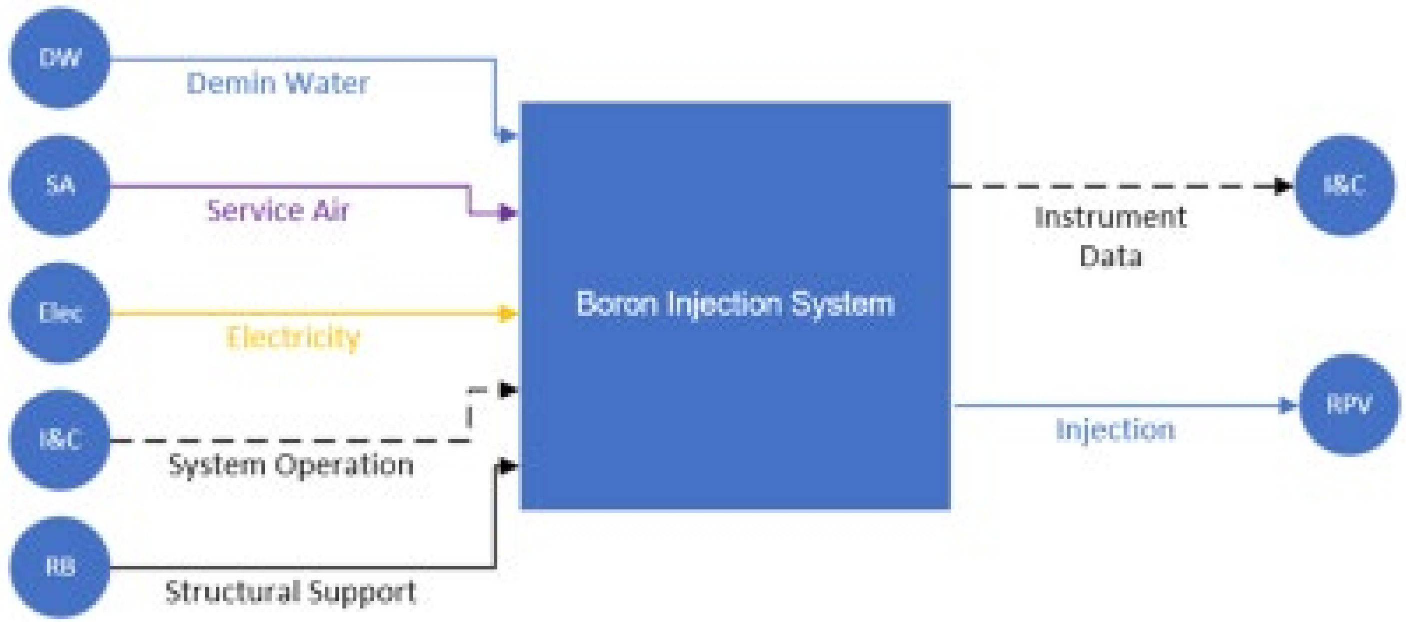

Figure 6-16: Boron Injection System Interfaces

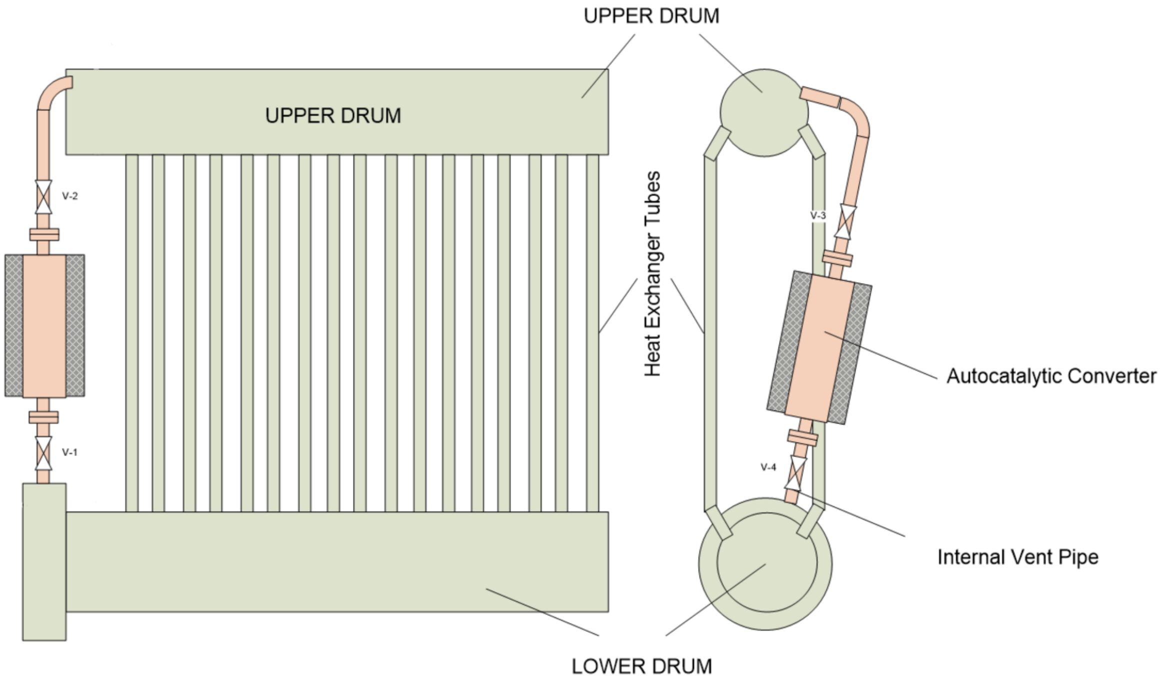

Figure 6-17: Preliminary Hydrogen Recombiner Configuration

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APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE AND ALARP

A.1 Claims, Argument, Evidence

The Office for Nuclear Regulation (ONR) "Safety Assessment Principles (SAPs) for Nuclear Facilities," (Reference 6-44), identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The Claims, Argument & Evidence (CAE) approach can be explained as follows:

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

The GDA CAE structure is defined within NEDC-34140P, "BWRX-300 Safety Case Development Strategy," (Reference 6-45), 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 Safety Case Development Strategy (SCDS) (Reference 6-45) and are as follows:

> *2.1.2: The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.*
>
> *2.1.3: The system/structure design has been undertaken in accordance with relevant design codes and standards (RGP) and design safety principles and taking account of Operating Experience to support reducing risks ALARP.*
>
> *2.1.4: System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning.*
>
> *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.*
>
> *2.1.6: The BWRX will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people.*
>
> *2.4.1: Relevant Good Practice (RGP) has been taken into account across all disciplines.*
>
> *2.4.2: Operational Experience (OPEX) and Learning from Experience (LfE) has been taken into account across all disciplines.*
>
> *2.4.3: Optioneering (all reasonably practicable measures have been implemented to reduce risk).*

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 A-1 below).

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.

A.2 Risk Reduction As Low As Reasonably Practicable

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 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:
  - Relevant Good Practice (RGP) has demonstrably been followed.
  - Operational Experience (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 6-45)

Consideration of the probabilistic safety aspects of the ALARP argument are out of the scope of this chapter.

In terms of PSR Ch. 6 ALARP aspects, the BWRX-300 ESFs include a passive reactor cooling system (ICS) that does not require external sources of power or operator actions to perform its safety functions. No immediate operator responses are required for safety, or to initiate shutdown as these are all automatically initiated. The ICS design is based upon decades of OPEX for similar BWR systems operations and also utilises ESBWR development technology experience.

The BWRX-300 has also introduced a dry containment enabled by the overall LOCA mitigation approach. The containment uses a nitrogen-inerted atmosphere which has been proven through OPEX from the ABWR. OPEX from the ABWR has also resulted in the design of the Passive Containment Cooling System which relies on natural circulation and condensation to remove heat without needing pumping or instrumentation to control or operate. OPEX also showed that the system should operate passively eliminating operator errors and mechanical failures. The significant reduction in active equipment has reduced the need for intrusive maintenance and operator dose uptake.

All of the above aspects are considered to contribute to reducing safety risks ALARP.

Table A-1: Claims, Arguments, Evidence Route Map

L3 No.	Level 3 Chapter Claim	Chapter 6 Arguments	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 (Engineering Analysis).			
2.1.2	The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.	Safety functions associated with the relevant SSC have been substantiated during normal operating conditions (including design codes and standards compliance).	6.1 Engineered Safety Feature Materials 6.2.1 Safety Design Bases (ICS) 6.2.1 Materials (ICS) 6.2.1 Performance and Safety Evaluation (ICS) 6.5.1 Containment Functional Requirements 6.5.2 Safety Design Bases (PCS) 6.5.6 Safety Design Bases and Materials (Containment Isolation) 6.6 Performance and Safety Evaluation (Control Room Habitability)
		A record of safe BWR plant operation and continuous improvement demonstrates a well-founded design.	NEDC-34137P, “BWRX-300 Design Evolution,” (Reference 6-46).
		Safety functions associated with the relevant SSC have been substantiated during hazard and fault conditions.	6.1 Engineered Safety Feature Materials 6.2.1 Safety Design Bases (ICS) 6.2.1 Materials (ICS) 6.2.1 Performance and Safety Evaluation (ICS) 6.5.1 Containment Functional Requirements 6.5.2 Safety Design Bases (PCS) 6.5.6 Safety Design Bases and Materials (Containment Isolation) 6.6 Performance and Safety Evaluation (Control Room Habitability) Note: Detailed design substantiation of safety functions is outside of the scope of a 2-Step GDA.

L3 No.	Level 3 Chapter Claim	Chapter 6 Arguments	Subsections and/or Reports that Evidence the Arguments
		Any shortfalls in safety function substantiation have been identified and assessed to identify any reasonably practicable means to reduce risk.	This argument is out of the scope of a 2-Step GDA and will be addressed during a site-specific stage (when more detailed evidence is developed).
2.1.3	The system/structure design has been undertaken in accordance with relevant design codes and standards (RGP) and design safety principles and taking account of Operating Experience to support reducing risks ALARP.	Design evolutions to SSC have been considered taking into account relevant BWR OPEX, and any reasonably practicable changes to reduce risk have been implemented.	NEDC-34137P, BWRX-300 Design Evolution (Reference 6-46).
		The SSC have been designed in accordance with relevant codes and standards (RGP).	006N3441, "BWRX-300 Applicable Codes, Standards, and Regulations List," (Reference 6-47). NEDC-34139, "BWRX-300 UK Codes and Standards Assessment," (Reference 6-48).
		The SSC have been designed in accordance with an appropriate suite of design safety principles.	The GEH Safety and Design Principles are documented in the 006N5064, "BWRX-300 Safety Strategy," (Reference 6-49). These principles are also presented within PSR Ch. 3.
2.1.4	System/structure performance will be validated by suitable testing throughout manufacturing, construction and commissioning.	SSC pre-commissioning tests (e.g., Non-Destructive Testing (NDT)) validate the relevant performance requirements.	This is considered to be beyond of the scope of a 2-Step GDA to define.
		SSC commissioning tests (e.g., system level pressure and leak tests) validate the relevant performance requirements.	This is considered to be beyond of the scope of a 2-Step GDA to define.
		SSC are manufactured, constructed and commissioned in accordance with QA arrangements appropriate to their safety classification.	DBR-0066822, "BWRX-300 System Functional Requirements (A11)," (Reference 6-50), describes how safety categorisation and SSC classification are linked to quality group (QA arrangement) definition. Reference 6-50 describes the high-level construction quality assurance and quality control arrangements and responsibilities.

L3 No.	Level 3 Chapter Claim	Chapter 6 Arguments	Subsections and/or Reports that Evidence the Arguments
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.	SSC ageing and degradation mechanisms will be identified during SSC design. These will be assessed to determine how they could potentially lead to SSC failure.	This is considered to be out of the scope of a 2-Step GDA, where the design maturing is at a concept stage. However, there is an intention to identify SSC ageing and degradation mechanisms, taking into account operational experience. Some early examples of how ageing and degradation have been addressed include materials selection, radiation shielding and water chemistry considerations.
		Appropriate Examination, Maintenance, Inspection and Testing (EMIT) arrangements will be specified taking into account SSC ageing and degradation mechanisms.	This is considered to be out of the scope of a 2-Step GDA, where the design maturing is at a concept stage. However, early project examples of such considerations are included within the following report: 006N6279, "BWRX-300 In Service Inspection Requirements," (Reference 6-51).
		The SSCs that cannot be replaced have been shown to have adequate life.	This is considered to be out of the scope of a 2-Step GDA, where the design maturing is at a concept stage.
		Ageing and degradation OPEX will be considered as part of the design stage component/materials selection process in order to mitigate SSC failure risk.	This is considered to be out of the scope of a 2-Step GDA, where the design maturing is at a concept stage. However, early project examples of such considerations are included within the following report: NEDC-34137P, BWRX-300 Design Evolution (Reference 6-46)

L3 No.	Level 3 Chapter Claim	Chapter 6 Arguments	Subsections and/or Reports that Evidence the Arguments
2.1.6	The BWRX will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people.	SSC decommissioning is considered at the design stage to ensure that safe decommissioning may take place.	OPEX demonstrates that decommissioning of reactor facilities is facilitated if the following are considered during the design phase: [1] Materials are selected to minimise the quantities of radioactive waste and assisting decontamination, [2] Plant layout is designed to facilitate access for decommissioning or dismantling activities, [3] Future potential requirements for storage of radioactive waste. See PSR Ch. 21 NEDC-34193P, "BWRX-300 Decommissioning and End of Life Aspects," (Reference 6-52).
		SSC are designed in order to minimise impacts on people and the environment during decommissioning.	See PSR Ch. 21: BWRX 300 Decommissioning planning (Reference 6-52).
2.4 Safety risks have been reduced as low as reasonably practicable.			
2.4.1	Relevant Good Practice (RGP) has been taken into account across all disciplines.	Relevant SSC codes and standards (RGP) are identified.	BWRX-300 Applicable Codes, Standards, and Regulations List (Reference 6-47). BWRX-300 UK Codes and Standards Assessment (Reference 6-48).
		SSC have been designed in accordance with relevant codes and standards (RGP).	BWRX-300 Applicable Codes, Standards, and Regulations List (Reference 6-47). BWRX-300 UK Codes and Standards Assessment (Reference 6-48). The descriptions included within this chapter identify how the SSC have been designed in accordance with relevant codes and standards.
		Any shortfalls in codes and standards compliance are identified and assessed to reduce risks ALARP.	Out of the scope of this PSR chapter.

L3 No.	Level 3 Chapter Claim	Chapter 6 Arguments	Subsections and/or Reports that Evidence the Arguments
2.4.2	Operational Experience (OPEX) and Learning from Experience (LfE) has been taken into account across all disciplines.	Design improvements to SSC have been identified considering relevant OPEX and LfE.	NEDC-34137P, BWRX-300 Design Evolution (Reference 6-46).
		Any reasonably practicable design changes to reduce risk have been implemented.	NEDC-34137P, BWRX-300 Design Evolution (Reference 6-46).
2.4.3	Optioneering (all reasonably practicable measures have been implemented to reduce risk).	Design optioneering has been performed in accordance with an approved process.	006N3139, "BWRX-300 Design Plan," (Reference 6-53).
		Design optioneering has considered all reasonably practicable measures.	006N3139, BWRX-300 Design Plan, (Reference 6-53). NEDC-34137P, BWRX-300 Design Evolution (Reference 6-46).
		Any reasonably practicable design changes to reduce risk have been implemented.	NEDC-34137P, BWRX-300 Design Evolution (Reference 6-46).

APPENDIX B FORWARD ACTION PLAN ITEMS

The FAP is not required to capture the 'normal business' of Safety, Security, Safeguards and Environmental case development as the design progresses from concept to design for construction and commissioning. FAP items can arise from several sources:

- Assumptions and commitments made in the GDA submissions that will require future verification/ implementation, for example, by the future constructor and/or plant operator,.A gap in the underpinning of the GDA submissions currently under development,
- A potential gap in a future phase of submissions if additional work is not performed.
- A gap identified by the regulators and communicated to the Requesting Party (RP) through a Regulatory Query (RQ) or Regulatory Observation (RO).

FAP items are included within the project's FAP report NEDC-34274P, "BWRX-300 UK GDA Forward Action Plan," (Reference 6-54), and its supporting commitments register.

There are no FAP items identified associated with PSR Ch. 6.