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# BWRX-300 UK Generic Design Assessment (GDA) Chapter 5 – Reactor Coolant System and Associated Systems

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# EXECUTIVE SUMMARY

The purpose of this Preliminary Safety Report (PSR) chapter is to describe the BWRX-300 Reactor Coolant System (RCS) and its associated systems, which generates and delivers steam to the turbine for power generation, and how it will comply with its design and safety requirements.

This chapter presents a level of detail commensurate with a 2-Step GDA and is structured in line with the high-level contents of SSG-61.

The scope of this chapter is the RCS and its Associated Systems. The RCS is comprised of the Safety Class 1 (SC1) portion of the Nuclear Boiler System (NBS) and the SC1 portion of the Condensate and Feedwater Heating System. The RCS extends to and includes the outermost Containment Isolation Valve in the Main Steam (MS) and Feedwater piping. The NBS and interfacing systems form the Reactor Coolant Pressure Boundary (RCPB). The Reactor Pressure Vessel (RPV) forms a major portion of the RCPB and contains the path for reactor coolant flow through the fuel to then generate steam.

This chapter describes the relevant systems within the RCS, their required safety and non-safety functions, design bases and provide arguments as to how the functions will be met by the systems. The functional aspects of the fuel and reactivity control within the RPV are discussed in PSR Ch. 4. The BWRX-300 design includes Engineered Safety Features (ESFs) that mitigate the consequences of Anticipated Operational Occurrences or postulated Design Basis Accidents without any core damage. The ESFs associated with the RCS interfacing systems are described in PSR Ch. 6, Section 6.2.

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

Claims, arguments, and evidence relevant to GDA step 2 objectives and scope are summarised in Appendix A, along with an ALARP position.

# ACRONYMS AND ABBREVIATIONS

| Acronym | Explanation |
|---|---|
| ALARP | As Low As Reasonably Practicable |
| ASTM | American Society for Testing and Materials |
| AOO | Anticipated Operational Occurrence |
| ASME | American Society of Mechanical Engineers |
| BIS | Boron Injection System |
| BPVC | Boiler and Pressure Vessel Code |
| BWR | Boiling Water Reactor |
| BWRVIP | Boiling Water Reactor Vessel and Internals Project |
| CAE | Claims, Arguments and Evidence |
| CFR | Code of Federal Regulations |
| CFS | Condensate and Feedwater Heating System |
| CIV | Containment Isolation Valve |
| CRD | Control Rod Drive |
| CUW | Reactor Water Cleanup System |
| D-in-D | Defence-in-Depth |
| DBA | Design Basis Accident |
| DCIS | Distributed Control and Information System |
| DEC | Design Extension Condition |
| DL | Defence Line |
| DL2 | Defence Line 2 |
| DL3 | Defence Line 3 |
| dP | Differential Pressure |
| ECCS | Emergency Core Cooling System |
| ESF | Engineered Safety Feature |
| FAC | Flow Accelerated Corrosion |
| FAP | Forward Action Plan |
| FMCRD | Fine Motion Control Rod Drive |
| FW | Feedwater |
| FWH | Feedwater Heater |
| FWRIV | Feedwater Reactor Isolation Valve |
| GALL | Generic Aging Lessons Learned |
| GDA | Generic Design Assessment |
| GEH | GE-Hitachi Nuclear Energy |
| HAZ | Heat Affected Zone |
| IASCC | Irradiation Assisted Stress Corrosion Cracking |
| I&C | Instrumentation and Control |

| Acronym | Explanation |
|---|---|
| IC | Isolation Condenser |
| ICS | Isolation Condenser System |
| IGSCC | Intergranular Stress Corrosion Cracking |
| ISI | Inservice Inspection |
| IST | In-Service Testing |
| $K_{IC}$ | Fracture Toughness |
| LfE | Learning from Experience |
| LOCA | Loss of Coolant Accident |
| MCR | Main Control Room |
| MPL | Main Parts List |
| MS | Main Steam |
| MSCIV | Main Steam Containment Isolation Valve |
| MSL | Main Steam Line |
| MSR | Moisture Separator Reheater System |
| MSRIV | Main Steam Reactor Isolation Valve |
| NBS | Nuclear Boiler System |
| NDE | Non-Destructive Examination |
| NR | Narrow Range |
| OLC | Operational Limits and Conditions |
| OLNC | On-Line NobleChem™ |
| OPEX | Operational Experience |
| PAS | Plant Automation System |
| PCS | Primary Containment System |
| PRNM | Power Range Neutron Monitoring System |
| PSR | Preliminary Safety Report |
| RB | Reactor Building |
| RCPB | Reactor Coolant Pressure Boundary |
| RCS | Reactor Coolant System |
| RG | Regulatory Guide |
| RGP | Relevant Good Practice |
| RIV | Reactor Isolation Valve |
| RPV | Reactor Pressure Vessel |
| RTP | Rated Thermal Power |
| $RT_{NDT}$ | Reference Nil-ductility Transition Temperature |
| SC | Safety Class |
| SC1 | Safety Class 1 |
| SC2 | Safety Class 2 |

| Acronym | Explanation |
|---|---|
| SC3 | Safety Class 3 |
| SCC | Stress Corrosion Cracking |
| SDC | Shutdown Cooling System |
| SDD | System Description Document |
| SIR | Seismic Interface Restraint |
| SSC | Structures, Systems, and Components |
| TAF | Top of Active Fuel |
| TASS | Turbine Auxiliary Steam Subsystem |
| TB | Turbine Building |
| TBV | Turbine Bypass Valve |
| TCV | Turbine Control Valve |
| TSV | Turbine Stop Valve |
| UK | United Kingdom |
| USNRC | U.S. Nuclear Regulatory Commission |
| UT | Ultrasonic Testing |
| WR | Wide Range |
| WRNM | Wide Range Neutron Monitor |

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| Revision # | Section Modified | Revision Summary |
|---|---|---|
| A | All | Initial Issuance |

# 5. REACTOR COOLANT SYSTEM AND ASSOCIATED SYSTEMS

The purpose of Preliminary Safety Report (PSR) Chapter 5 is to describes the BWRX-300 RCS, which generates and delivers steam to the turbine for power generation, and how it will comply with its design and safety requirements. The BWRX-300 is a natural circulation reactor and therefore does not rely on pumps to drive core coolant flow. The thermal-hydraulic design of the RPV for natural circulation is described in NEDC-34166P, "BWRX-300 UK GDA Ch. 4: Reactor," (Reference 5-1), Section 4.4.

The RCS and interfacing systems form the RCPB. The RPV forms the major portion of the RCPB and contains the path for reactor coolant flow through the fuel and generates steam.

PSR Ch. 5 presents a level of detail commensurate with a 2 Step Generic Design Assessment (GDA) and is structured in line with the high-level contents of IAEA SSG-61.

The scope of PSR Ch. 5 is the Reactor Coolant System and Associated Systems. This chapter does not include any design code demonstrations for the major components or any detailed design substantiation.

This chapter describes the following RCS aspects:

- Summary of RCS
- Materials used
- Associated RCS systems
- RPV, piping, component supports and restraints
- Reactor pressure control system
- Interfacing and auxiliary systems

PSR Ch. 5 describes the relevant systems within the RCS, their required safety and non-safety functions, design bases and provide arguments as to how the functions will be met by the systems.

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

The RCS includes the components necessary to provide and maintain adequate core cooling conditions (i.e., pressure, temperature, and coolant flow rate) for the fuel in power operation. The RCS is comprised of the SC1 portion of the BWRX-300 NBS and the SC1 portion of the Condensate and Feedwater Heating System (CFS). The RCS extends to and includes the outermost Containment Isolation Valve (CIV) in the MS and Feedwater (FW) piping (Figure 5-2).

The NBS includes the RPV and provides MS flow paths from the RPV to the steam turbine, and the CFS provides FW flow paths to the RPV. The NBS, up to and including the outermost Main Steam Containment Isolation Valves (MSCIVs), is depicted in Figure 5-2.

The RCPB is comprised of the pressure retaining components of the RCS and connected systems within the Primary Containment System (PCS) up to and including their respective outboard CIVs. The RPV forms the major portion of the RCPB and contains the path for reactor coolant flow through the fuel and generates steam. The connected systems within the PCS or partially within the PCS in the case of the Isolation Condenser System (ICS), include the following:

- The ICS which connects to the RPV
- The Reactor Water Cleanup System (CUW) which connects with the RPV

- The Boron Injection System (BIS) which interfaces with ICS for the flow path to the RPV

The Isolation Condensers (IC) which are ICS components, are in the Reactor Building (RB) directly above the PCS. The ICS is described in PSR Ch. 6, Section 6.2.1. The Shutdown Cooling System (SDC), which interfaces with the ICS and CFS outside the PCS for the flow path with the RPV, is described in PSR Ch. 9A, Section 9A.2-5. The CUW is described in Section 5.1.8 and PSR Ch. 9A, Section 9A.2.2. The BIS is described in PSR Ch. 9, Section 9.3.10.

Further, this chapter addresses the functional and structural integrity aspects of the various NBS Structures, Systems, and Components (SSC) that are designed with robustness, quality, independence, redundancy, and diversity to maintain adequate reactor coolant inventory during Normal Operation, Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Design Extension Conditions (DECs). The NBS and RCPB design utilise the Safety Strategy Defence-in-Depth (D-in-D) Defense Lines (DLs) as described in NEDC-34165P, "BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs," (Reference 5-2), Section 3.1 to meet international regulatory requirements and expectations. The BWRX-300 approach to classifying of SSC is discussed in PSR Ch. 3, Section 3.2.

The portion of the NBS and system interfaces located in the RB is illustrated in Figure 5-2 and described in this chapter. The portion of NBS located in the Turbine Building (TB) is presented in Figure 5-3 and discussed in detail in Section 10.1 of NEDC-34174P, "BWRX-300 UK GDA Ch. 10: Steam and Power Conversion Systems," (Reference 5-3).

The functional aspects of the fuel and reactivity control within the RPV are discussed in PSR Ch. 4. The BWRX-300 design includes ESFs that mitigate the consequences of AOOs or postulated DBAs without core damage. The ESFs associated with the NBS interfacing systems are described in Section 6.2.1 NEDC-34168P, "BWRX-300 UK GDA Ch. 6: Engineered Safety Features," (Reference 5-4), Section 6.2.1. Additional structural integrity requirements to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) requirements for High Integrity, Class 1, Class 2, and Class 3 metallic components are described in NEDC-34194P, "BWRX-300 UK GDA Ch. 22: Structural Integrity," (Reference 5-5).

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 has been included to capture any Forward Action Plan (FAP) items. However, none were identified for PSR Ch. 5.

## 5.1 Summary Description

The BWRX-300 RCS is comprised of three primary subsystems: RPV, MS, and RCS instrumentation. In addition to these subsystems, the RCS also provides RPV head venting and RPV head flange seal leak detection features. Within the NBS, the SC1 Reactor Isolation Valves (RIVs) are mechanically redundant and Instrumentation and Control (I&C) diverse. These RIVs interface with the following subsystems and systems: MS, RPV Head Vent, ICS Supply, ICS Return, CFS, and CUW. The Reactor Coolant System and associated systems are designed in accordance with requirements of ASME BPVC-III NB, "Section III- Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," (Reference 5-6) and ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components," (Reference 5-7).

The BWRX-300 NBS and interfacing systems are a unique design which mitigates Loss of Coolant Accidents (LOCAs) that could impact Reactor Vessel Water Levels by providing double RIVs directly flanged to the integral RPV nozzles. The RIVs are closer to the RPV compared to predecessor Boiling Water Reactor (BWR) designs (i.e., there is no piping between the RPV and RIVs). Figure 5-1 provides the input-output block diagram that lists all intersecting systems with the NBS and describes the interfaces between systems. A breakdown of the Main Parts List (MPL) codes associated with Figure 5-1 is provided in Table 5-1. The portion of the NBS (Main Steam Section) from the Seismic Interface Restraint (SIR) to the Turbine Stop Valves (TSV), Turbine Bypass Valves (TBV), and other steam loads is addressed in PSR Ch. 10, Section 10.1.

The NBS is built on utilisation of inherent margins (e.g., larger water inventory) to eliminate system challenges, and reduce the number and size of RPV nozzles as compared to predecessor designs. All nozzles are located significantly above the Top of Active Fuel (TAF). The relatively large RPV volume, along with the relatively tall chimney region, provides a substantial reservoir of water above the core. This ensures the reactor water level is maintained above TAF and fuel cladding temperature is maintained within normal operating temperature range following transients involving FW flow interruptions or LOCAs. These design safety features preserve reactor coolant inventory to ensure that adequate core cooling is maintained.

Innovative features related to BWRX-300 are the RPV chimney region flow characteristics (see Section 5.4); RIV qualification (see Section 5.10.7) and the RPV water level measurement in the core region (see Section 5.3.1). The large RPV volume enhances safety by reducing the rate at which reactor pressurisation occurs if the reactor is suddenly isolated from its normal heat sink. The containment isolation function is addressed separately in PSR Ch. 6, Section 6.5.

Figure 5-2 provides the portion of the NBS that is described in this chapter and shows the double isolation RIVs of those systems that intersect the RPV. Additionally, Figure 5-4 provides the extent of the RCPB.

### 5.1.1 Reactor Coolant Pressure Boundary

The following information is consistent with 006N7828, “BWRX-300 Nuclear Boiler System, System Description Document (SDD),” (Reference 5-8). The BWRX-300 RCPB is designed in accordance with ASME BPVC to maintain its integrity for the design life of the plant. The RCPB is consistent with the definition provided in 10 Code of Federal Regulations (CFR) 50.2 and includes the pressure retaining components of the RCS identified in Figure 5-2.

The RCPB (see Section 5.3), contains all pressure-retaining components such as:

- The RPV
- The RIVs
- The piping from the outboard RIVs up to and including each respective outboard CIV (this includes the redundant isolation valve downstream of the CIV connected to the ICS to SDC interface piping)
- The ICS steam supply lines, isolation condensers, purge lines, and condensate return lines
- The head vent piping to the Main Steam Line (MSL) and head vent interface piping to quench tank up to and including the isolation valves for the quench tank
- The reactor instrumentation lines
- The Control Rod Drive (CRD) interface flanges/housings
- The nuclear instrumentation interface flanges/housings

The RCPB is presented in Figure 5-4.

### 5.1.2 Reactor Pressure Vessel and Appurtenances

The RPV is used as a pressure-retaining barrier of primary coolant in a BWRX-300 reactor. The vessel contains the light water coolant/moderator and forms the path for core recirculation flow. The vessel also contains nuclear fuel, saturated steam, fuel supporting structures, and internals required for safe operation of the reactor. Production of thermal energy from the nuclear fission process takes place within the vessel. The above function requires that the vessel be a SC1 vessel, designed to meet the requirements of ASME BPVC, Section III, Division 1, Subsection NB (see Figure 5-6 the major components contained in the RPV). A more detailed description of the RPV and its major components is provided in Section 5.4.

The RPV design is such that the physical level of all nozzles are located significantly above TAF.

The RPV has isolation valves attached directly to the RPV integral nozzles via flanged bolted connections. The SC1 RIVs are mechanically redundant and have diverse I&C. In addition, a flanged connection is provided in the RPV bottom head for each Fine Motion Control Rod Drive (FMCRD) and each neutron monitoring detector.

Table 5-2 includes materials and specifications for the RPV and RPV appurtenances which form part of the RCPB. This includes RPV nozzles, stub tubes, CRD flanges/housings, and in-core instrumentation flanges/housings and their associated RCPB components.

The RPV head vent subsystem (SC1) includes piping internal to the RPV head, two RIVs in-series flange-mounted on the reactor, and piping to one of the MS lines and the Quench Tank, both connections located inside the PCS.

The major safety consideration for the RPV is the ability of the vessel to function as a radioactive material barrier under Normal, AOO, DBA, and DEC plant states as described in PSR Ch. 3, Section 3.1.

### 5.1.3 Main Steam Subsystem

The MS subsystem consists of two steam lines from the discharge flange of the outboard Main Steam Reactor Isolation Valves (MSRIVs) to the TSVs, the TBVs, the MSL drains, and other load isolation/maintenance valves. The supply lines to these loads, all connecting branch lines up to and including their respective isolation valves, and all associated piping supports are also part of the MS subsystem.

The MSL drains remove any condensate from the MSLs to the main condenser during startup, low power operation, normal power operation, and shutdown. A reduction in power to a low-level, results in the automatic opening of the air-operated drain line valves, thereby establishing drain flow to the main condenser.

The portion of the MS piping from the outboard MSRIVs to the SIR outboard of the MSCIVs is SC1 and designed to meet the requirements of ASME BPVC, Section III, Division 1, Class 2. The remaining MS piping from the SIR to the TSVs, the TBVs, the MSL drains, and other load isolation/maintenance valves is designed to meet the requirements of ASME B31.1, “Power Piping,” (Reference 5-9) as described in Sections 10.1, 10.2 and 10.4 of NEDC-34174P (Reference 5-3).

### 5.1.4 Nuclear Boiler System Instrumentation Subsystem

The NBS instrumentation subsystem is the source of the reactor water level, reactor pressure, temperature, and steam flow information. The NBS instrumentation consists of sensors to measure and monitor parameters including reactor coolant pressure, flow, temperature, water level, and reactor power. Additionally, there are sensors to measure and monitor MSL pressure, flow, drain temperature, RPV metal temperatures, RPV head flange seal pressure (leak detection), RPV head venting, steam tunnel temperature, and TB temperatures near the MSLs for leak detection purposes.

Neutron flux detectors for reactor power provide signals to the Power Range Neutron Monitoring System (PRNM) and the Wide Range Neutron Monitoring System (WRNM) (refer to Section 7.3.1 of NEDC-34169P, "BWRX-300 UK GDA Ch. 7: Instrumentation and Control," (Reference 5-10), Section 7.3.1 (System design bases and associated safety functions) for detailed discussion). The non-neutron flux sensors provide data to instruments or signal conditioning and data acquisition equipment that are typically mounted on instrument racks outside of containment. The instruments are appropriately separated and divided into DLs and safety classifications.

The NBS also has control requirements for valves (RIVs, CIVs, and drain valves) that are repositioned as the plant is brought from cold to full power or that respond to unplanned transients. Control design requirements include that SC1 equipment is optically isolated from the SC3 Plant Automation System (PAS). Therefore, provisions are made for the reactor operator to manually operate the valves as prompted by the PAS or Distributed Control and Information System (DCIS) Visual Display Units. The SC3 valves may be operated by the PAS.

The nuclear boiler controller logic architecture (see PSR Ch. 7, Section 7.2.1) uses Triple Modular Redundant technology and is dual ported to the plant nuclear segment network. As with all Triple Modular Redundant controllers, extensive hardware and software diagnostics are provided to provide for operator monitoring and alarms. There is no connection between Defence Line 3 (DL3)/SC1, and the nuclear segment controllers and no sensors or actuators are in common.

**Reactor Pressure Vessel Level Instrumentation**

Normal operations Defence Line 2 (DL2) for RPV water level are controlled by FW makeup flow control. Water level measurement by Differential Pressure (dP) is also used for transient/AOO (DL2) and DBA (DL3) monitoring. The dP instrumentation in predecessor plants have been setup to provide a more accurate NR measurement as well as a less accurate, but otherwise sufficient WR measurement.

The BWRX-300 is designed to have no RPV fluid nozzles below TAF + 4 m. This is a plant design commitment to preserve sufficient condensate inventory for analyzed events and accidents. Because there are not any operator actions dependent on direct water level measurement between TAF and TAF + 4 m, there are no water level measurements available in this range.

The water inventory in the RPV is credited for normal operations and AOOs as described in PSR Ch. 15, Section 15.5.3 (decrease/increase in reactor coolant inventory) of NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analyses," (Reference 5-11). The reactor level control remains unaffected by these failures and is able to maintain level.

**Reactor Coolant System Pressure, Flow, and Temperature**

The other NBS instrumentation functions are to measure reactor coolant pressure, flow, and temperature. More details are provided in Section 5.3.1 (NBS instrumentation subsystem).

### 5.1.5 Main Steam Flow Restrictors

The MS flow restrictors are located in the RPV MS nozzles. They perform functions in various safety categories and corresponding defence levels. The Safety Category 1 functions are to limit the overall steam flow through one steam line when the steam flow exceeds preselected operational limits (if a large break of a MSL is sensed) and the corresponding MSRIVs and MSCIVs have not closed. The other Safety Category 1 function is to provide that same sensing function to the SC1 MSRIV and MSCIV isolation system. The Safety Category 2 steam flow function also supplies the same sensing functions for MSRIV and MSCIV isolation in addition to the already discussed Safety Category 1 isolation function. The Safety Category 3 steam flow function supports the overall reactor water level controller that maintains target water level as one of the inputs to that controller.

The MS flow restrictor provides low-pressure sensing taps for instrumentation lines for dP transmitters that are used to measure steam flow. MS flow is measured through the MS nozzle venturis using Bernoulli's equation of dP to derive the MS mass flow.

The MS flow restrictor is designed in accordance with the Report of ASME Research Committee on Fluid Meters and has no moving parts. Required plant duty cycles associated with MS flow restrictors and other ASME BPVC, Section III, Division 1, Class 1 components are provided in Section 3.6 of Attachment 1 in NEDC-34165P (Reference 5-2).

### 5.1.6 Reactor Isolation Valves and Main Steam Containment Isolation Valves

A feature of the BWRX-300 are the RIVs [004N9515 "BWRX-300 Reactor Vessel Integral Isolation Valve Basis," (Reference 5-12) and 006N6121, "Reactor Pressure Vessel and Containment Isolation Valves," (Reference 5-13)] that are directly flanged to nozzles on the reactor vessel instead of welding a pipe to the vessel nozzle and then connecting the isolation valve to the pipe at some distance from the reactor. As appurtenances to the RPV, the RIVs allow large pipe break LOCAs to be quickly isolated, thereby, retaining the fluid inventory in the vessel for core cooling. Each RPV nozzle with an internal bore greater than DN20 is required to have two RIVs flanged directly to the nozzle. The main function of the isolation valves is to close and mitigate the effects of LOCAs from large and medium pipe breaks. The RPV automatic isolation concept consists of two RIVs in series and is single failure proof with each of the RIVs independently able to isolate the line. RIVs are placed inboard of each large and medium sized pipe that is connected to the RPV.

The BWRX-300 design includes one MSCIV for each MS line outside the containment. If a MS line break occurs inside the containment, closure of all outside CIVs seals the containment.

Additionally, all RIVs with the exception of the Isolation Condenser Supply Reactor Isolation Valves (ICSRIVs) and Isolation Condenser Return Reactor Isolation Valves (ICRRIVs) will automatically close to isolate the RCPB at the RPV when a pipe break occurs either inside or outside of containment. This action limits the loss of reactor coolant and the release of radioactive materials. Further discussion on operation of the RIVs and MSCIVs is provided in Section 5.10.

### 5.1.7 Control Rod Drive System

The Control Rod Drive (CRD) contains components which form part of the RCPB, along with components which are important to safety to shut down the reactor. Those portions of the CRD are classified as DL3/SC1.

The CRD housing (attached to the RPV), the FMCRD middle flange including the ball check valve, and the lower component housing, which enclose the lower part of the drive, are all part of the reactor pressure boundary. The middle flange is attached to the CRD housing by four threaded bolts. The lower housing is, in turn, held to the middle flange and secured to the CRD housing by a separate set of eight main mounting bolts that become a part of the reactor pressure boundary. This arrangement permits removing the lower housing without disturbing the rest of the drive. Removing the lower component housing transfers the weight of the drive line from the drive shaft to a seat in the middle flange. Both the ball screw and drive shaft are locked to prevent rotation while the two are separated.

The part of the drive inserted into the CRD housing is contained within the outer tube. The outer tube is the drive hydraulic scram pressure boundary, eliminating the need for designing the CRD housing for scram pressure. The outer tube is welded to the middle flange at the bottom and is attached at the top with the CRD blowout support, which bears against the CRD housing. The blowout support and outer tube are attached by a slip-type connection that accounts for any slight variation in length between the drive and the CRD housing.

Purge water continually flows through the drive. The water enters through the ball check valve in the middle flange and flows around the hollow piston into the reactor. O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. The labyrinth seal is also designed to provide sufficient resistance during hydraulic scram to restrict flow from bypassing the hollow piston into the RPV.

The CRD system is further discussed in PSR Ch. 4, Section 4.5.

### 5.1.8 Reactor Water Cleanup System

The CUW system provides blowdown-type cleanup flow for the RPV during the reactor power operating mode. CUW also provides an overboarding flow path (i.e., excess RPV coolant inventory discharge for level control) to the condenser hotwell or Liquid Waste Management System directly from the RPV lower region.

Detailed design description of CUW is provided in PSR Ch. 9, Section 9A.2.2 of NEDC-34171P, "BWRX-300 UK GDA Ch. 9A: Auxiliary Systems," (Reference 5-14).

### 5.1.9 Isolation Condenser System

The ICS removes decay heat after any reactor isolation and shutdown event during power operations when the main condenser is not available. The ICS is connected to the RPV by steam supply piping and condensate return piping. The large ICS heat removal capacity limits increases in RPV pressure and maintains the RPV pressure at an acceptable level. The ICs condense steam from the RPV and transfer thermal energy by convection to the IC pool water.

The ICS is responsible for protecting the integrity of the RCPB through provision of decay heat removal and over-pressure protection in the event of a large line break LOCA that results in the isolation of the RPV.

During shutdown modes, ICS will perform the RPV isolation function for a line break within the SDC system by isolating the ICS-SDC interface valves. This function prevents an ICS train from being rendered inoperable by what otherwise would be an RIV closure in the case of an SDC LOCA. As described in PSR Ch. 15, Section 15.5.4 (Loss-of-coolant accidents - DBAs), ICS removes decay heat generated in the core and does not require coolant injection to mitigate pipe breaks and transients.

The ICS is an Emergency Core Cooling System (ECCS), and also performs the RPV and RCPB overpressure protection function for which further details can be found in PSR Ch. 6, Section 6.2.1.

### 5.1.10 Piping and Instrumentation Schematics

Piping and instrumentation schematic diagrams covering the systems included within NBS and connected systems are presented as follows:

- Reactor Coolant System (Figure 5-2 and Figure 5-3)
- Isolation Condenser System (Figure 6-3, NEDC-34168P)
- Reactor Water Cleanup System (Figure 9A.2-2, NEDC-34171P (Reference 5-14)
- 006N7708, "BWRX-300 Shutdown Cooling System" (Reference 5-15) (Section 9A.2.3 and Figure 9A.2-3 of NEDC-34171P)

#### 5.1.11 Elevation Schematics

The elevation schematic showing the principal features of the reactor and connecting systems in relation to the containment are provided in Figure 1-2 of NEDC-34163P, "BWRX-300 UK GDA Ch. 1: Introduction," (Reference 5-16).

## 5.2 Materials

The material and process control requirements for the BWRX-300 components have been defined to ensure the reliability of the plant operations through its design life, by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials, specifically from Intergranular Stress Corrosion Cracking (IGSCC) through material chemistry, heat treatment processes, processes contamination controls, and material processes controls.

The BWRX-300 NBS employs proven BWR materials and processes which have been refined to meet reactor requirements. BWRX-300 materials provide adequate strength, Fracture Toughness ($K_{IC}$), fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful Operational Experience (OPEX) in reactor service.

The basic design principle for materials selection is for the materials to be capable of maintaining reliable operation of plant systems and equipment during the design life of each component. In principle, carbon steels including atmospheric corrosion resistant carbon steels and low alloy steels are used as basic materials. In order to minimise levels of radiation from corrosion products, reactor internals are made of austenitic stainless steel. Materials selection and controls address materials degradation issues in the reactor system such as Stress Corrosion Cracking (SCC), general corrosion, and Flow Accelerated Corrosion (FAC).

SCC is considered the dominant form of corrosion in a BWR. Significant efforts through the years have been expended to understand it and control SCC. It is a complex phenomenon that involves mechanical, electrochemical, and metallurgical factors. SCC is usually characterised by localisation of the cracked region near welds or near regions of high surface strains or stresses. For SCC to occur, it requires three necessary components to be simultaneously present. The elimination of any one of these factors, or reduction of one of these factors below a threshold level eliminates the risk of SCC. These three necessary conditions for SCC as schematically shown in Figure 5-5 are (1) Susceptible material; (2) Tensile Stress (applied or residual) and (3) Corrosive environment.

The different degradation mechanisms that potentially affect the integrity of the construction materials that are used in the BWRX-300 are discussed in the subsequent sections. These construction materials were selected because they provide adequate strength, $K_{IC}$, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful OPEX in reactor service.

### 5.2.1 Basis of Material Selection and Component Fabrication

The structural materials used for the RPV and its attachments, the core support structure, internals, and the piping in the BWRX-300 design are selected and fabricated to meet the design requirements of the reactor components while maintaining high resistance to known degradation mechanisms during plant operation. Seals, gaskets, and fasteners are selected based on experience and industry acceptance. To monitor material degradation, guidelines for surveillance programs for BWRX-300 are informed by ASME and industry guidelines.

The BWR uses high purity water as the fluid to transfer thermal energy. The quality of water to which the component is exposed, to a large extent, determines the reliability and lifetime of that component. Generally, corrosion effects stem from the combination of materials selected, the stresses or duty applied, and the environment to which the component is exposed. The extent of materials corrosion from its environment is a function of impurity and its concentration, temperature, and time. Water of required quality is achieved by the use of appropriate water treatment equipment and systems, as well as the application of appropriate materials in system construction.

The chemistry regime at a nuclear power plant controls the reactor water chemistry in order to minimise SCC and radiation buildup in the plant's austenitic structural materials. BWRs were originally designed to operate with the chemistry controlled by the CFS and CUW without chemical additives. In the decades of OPEX since then, advancements have been made in understanding and improving water chemistry effects by controlling the chemistry via chemical additions. The planned BWRX-300 Chemistry Regime includes the following:

- Hydrogen Water Chemistry
- On-Line NobleChem™ (OLNC)
- Zinc Injection using Depleted Zinc Oxide.

Additionally, Hydrogen Water Chemistry in conjunction with noble metal technology, e.g., NobleChem™ or noble metal alloys, provides the ability to achieve Hydrogen Water Chemistry benefits while limiting radiation dose in the steam lines.

The BWRX-300 builds on a long history of material experience and refinement of material alloy chemistry to assure the best characteristics to preclude long-term material degradation. These materials include wrought alloys as well as weld metals used in the construction. Given the decades of experience as well as the strength of utility interactions through Electric Power Research Institute and research efforts by GE-Hitachi Nuclear Energy (GEH) and its licensees, there is high confidence that degradation will be very limited if not eliminated for the life of the plant. In addition, water chemistry environment controls to address potential degradation mechanisms are being implemented.

Section 5.2.3 covers low alloy steels and carbon steels with the associated weld metals used for the RPV, piping and vessel attachments (including nozzles and bolting). Section 5.2.4 includes a discussion of the potential degradation mechanisms. Section 5.2.5 discusses austenitic stainless steel and nickel-based alloys used for reactor internal components such as the core shroud, chimney, core support structure and control rod and instrumentation hardware as well as several piping systems. Relevant degradation mechanisms are discussed in Section 5.2.6, in particular SCC are discussed for these austenitic components.

All pressure boundary material specifications used in the BWRX-300 are as per American Society for Testing and Materials (ASTM) /ASME standards. Pressure boundary materials are selected with consideration to environment compatibility with an emphasis for limiting the effects of corrosion in the reactor coolant environment due to contaminants and radiolytic products. This is especially true for ferritic low alloy steels and carbon steels.

In the case of stainless steels, the major environment-related materials performance issue encountered to date in the RCPB of BWRs has been Intergranular IGSCC of sensitised austenitic stainless steel. IGSCC has been discovered in sensitised material adjacent to welds in Type 304L and Type 316L stainless steel piping systems. Substantial research and development programs have been undertaken to understand the IGSCC phenomenon. Remedial measures have been developed and implemented. for the BWRX-300. IGSCC resistance is achieved through the use of IGSCC resistant materials such as Type 316L stainless steel with controlled composition. Welding controls are also implemented to ensure that welding filler materials, welding techniques and controlled post-weld heat treatment are utilised to minimise the potential for IGSCC. Additionally, weld locations in the reactor internals welds are strategically placed to limit further sensitisation from irradiation. Water chemistry is also tightly controlled to minimise the existence of an environment that could promote IGSCC.

The BWRX-300 water chemistry program follows the best industry water chemistry guidelines and it is designed to analyze and monitor system chemistry for trending with alarm notification so actions can be taken to stay within operating specifications. This supports minimizing corrosion from chemical contaminants and monitoring chemical additives used to limit corrosion and radiation buildup. Details on the water chemistry program adopted for the BWRX-300 can be found in PSR Ch. 23, Section 23.1 of NEDC-34195P, “BWRX-300 UK GDA Ch. 23: Reactor Chemistry,” (Reference 5-17).

Different materials are used to manufacture the reactor components, the RPV, piping, and reactor internals. Additionally, the typical materials used for gaskets, seals, and fasteners will vary dependent on the location and environmental conditions. For the primary components, the key materials, typical composition limits and ASME/ASTM specifications along with the fabrication processes are used to meet the requirements for the BWRX-300 application to assure resistance to different types of degradation. GEH specified materials and process controls are used to define the requirements related to service conditions. The ASME Code specifications and material alloy chemistry, define their chemical, physical and mechanical properties, that produce resistance to corrosion, dimensional stability, and desired toughness in the fabricated components. Key microstructural features resulting from the material fabrication processing methods are also critical to achieving resistance to degradation over the plant lifetime.

#### 5.2.2 Material Properties

Nuclear power plant components may be exposed to environmental factors whose single and combined cumulative effects cannot be predetermined for the operating lifetime of the plant. The most important environmental factors are stress, high temperature, irradiation, operating transients, hydrogen absorption, corrosive attack, vibration, and fretting, with all these factors depending on time and operating history. These environmental factors can result in changes to material properties caused by irradiation or thermal embrittlement, corrosion fatigue and potentially lead to the initiation and growth of flaws. Therefore, the $K_{IC}$ properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness to minimise the possibility of brittle fracture of the NBS and RCPB components.

Materials used in the construction of the portion of the NBS that provide a safety function or materials that are in contact with the process fluid, are selected based on suitability for the service based on the requirements of the applicable codes and standards. NBS piping is designed using materials selected to prevent erosion caused by high local steam flow velocity.

Materials used for the PCS, penetration piping and the associated supports are designed in accordance with the rules and requirements of ASME BPVC, “Section II, Materials,” (Reference 5-18).

Where applicable, materials are impact tested in accordance with ASME BPVC, Section III, Division 1, Class 1, and Class 2 components, using instruments and equipment certified to meet the same ASME BPVC requirements.

The RPV also complies with the toughness requirements for Class 1 vessels per ASME BPVC, Section III. This includes plates, forgings, weld material, and the heat affected zones. Minimum toughness values at various locations of the RPV are confirmed through Charpy V-notch testing, as per ASME BPVC, Section III. Charpy curves and $K_{IC}$ specimens are used for evaluation of radiation effects on embrittlement with operating time. Consideration has been given to the effects of irradiation on beltline $K_{IC}$ by controlling the chemical composition of vessel beltline materials.

SC pressure boundary components are designed and manufactured to ASTM/ASME fabrication and material specifications. This ensures that the elastic analysis methods of ASME BPVC, Section III, including stress intensity factors are valid and offer a high degree of conservatism.

The surveillance test program specimens are used to monitor the radiation-induced changes in the mechanical properties of the core beltline region materials of the RPV. Part of the specimens that are exposed to irradiation are installed in removable specimen capsules at the inside vessel wall opposite the active core. The remaining unirradiated baseline specimens are used to establish material reference data.

The impact of irradiation on ferritic pressure boundary materials is described in ASME Section III, Non-mandatory Appendix W, Paragraph W-3300. As part of the design, vessel irradiation (neutron irradiation) is calculated over the 60-year design life of the plant. Because the core is the source of the neutrons, the fluence will be at a maximum in the region surrounding the active core, which is consistent with all previous BWR designs.

Surveillance specimen materials are prepared from samples are taken from the actual materials used to fabricate the beltline of the RPV. The weld and Heat Affected Zone (HAZ) samples are supplied if they are exposed to neutron fluence 1.0E17 n/cm$^2$ over the design life of 60 years.

The predicted changes from the initial properties are a function of chemical composition and the neutron fluence during reactor operation. The base and weld material with the highest predicted adjusted reference temperature at end of life are selected for the surveillance program. The prediction of adjusted reference temperature shift is in accordance with peak neutron fluence value specified in the vessel irradiation data drawing.

### 5.2.3 Overview of Reactor Pressure Vessel and Pressure Boundary Components

The RPV and associated core support structures and piping are designed to:

- Withstand static and dynamic loading, including thermal expansion and contraction
- Withstand vibration (such as flow induced and acoustic vibration)
- Ensure chemical compatibility, including service-related contaminants
- Meet thermal material limits
- Meet radiation damage limits

The primary materials used in the RPV and pressure boundary components are listed in Table 5-2, together with the applicable specifications. The RPV materials comply with the provisions of ASME BPVC, Section III, and shall also meet the requirements of ASME BPVC, Section II (Reference 5-18). The RPV materials also meet the additional requirements to address potential degradation mechanisms. These materials provide adequate strength, $K_{IC}$, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful operating experience in reactor service.

**Reactor Pressure Vessel and Nozzles**

The RPV is constructed from low alloy, high strength steel forgings or plates. Forgings are ordered to ASME material specification SA-508M, Grade 3, Class 1, while plates are ordered to ASME SA-533M, Type B, Class 1. This material is melted to fine grain practice and is supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels. Table 5-2 lists the specifications for the RPV Vessel and Nozzle materials. Nominal compositions for these low alloy steel and carbon steel materials used in reactor components are presented in Table 5-3. The details of all smaller components such as low alloy steel studs, nuts, and other components are consistent with the material in Table 5-2. Welding electrodes for low alloy steel are low hydrogen type ordered to ASME SFA-5.5, and weld filler metal to SFA-5.23 and SFA5.28. All forgings and bolting are 100 percent ultrasonically tested and surface examined as required by ASME BPVC, Section II (Reference 5-18). $K_{IC}$ properties of materials are also measured and controlled in accordance with ASME BPVC, Section III, Division 1 as discussed in Section 5.2.4.

The BWRX-300 RPV meets the requirements of:

- "10 CFR Appendix A to Part 50 - General Design Criteria for Nuclear Power Plants," (Reference 5-19), GDC 1 and GDC 30, as they relate to quality standards for design, fabrication, erection, and testing of SSCs, by compliance with ASME BPVC Section III, Section IX, "Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators," (Reference 5-20), and Section XI, "BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light Water-Cooled Plants," (Reference 5-21) of the ASME BPVC and by conformance with Regulatory Guide (RG) 1.31, RG 1.50, RG 1.65, Generic Letter 88-01 and NUREG-0313, Revision 2.
- 10 CFR 50, Appendix A, GDC 4, as it relates to compatibility of components with environmental conditions, by conformance with RG 1.44 and Generic Letter 88-01 and NUREG-0313, Revision 2.
- 10 CFR 50, Appendix A, GDC 14, as it relates to prevention of rapidly propagating fractures of the RCPB, by compliance with 10 CFR 50, Appendix G and conformance with RG 1.31 and Generic Letter 88-01 and NUREG-0313, Revision 2.
- 10 CFR 50, Appendix A, GDC 31, as it relates to material $K_{IC}$, by compliance with 10 CFR 50, Appendix G, and conformance with RG 1.65.
- 10 CFR 50, Appendix A, GDC 32, as it relates to the requirements for a materials surveillance program, by compliance with "Appendix H to Part 50 - Reactor Vessel Material Surveillance Program Requirements," (Reference 5-22) and ASTM International (ASTM) E185-21 "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," (Reference 5-23).
- 10 CFR 50.55a as it relates to quality standards for design and determination and monitoring of $K_{IC}$, by compliance with "Appendix G to Part 50 - Fracture Toughness Requirements," (Reference 5-24), Section III, Section IX, and Section XI of the ASME BPVC, and conformance with RG 1.31, RG 1.50, RG 1.65, Generic Letter 88-01 and NUREG-0313, Revision 2.

**Alloy Steel, Stainless Steel and Carbon Steel Piping**

The information for the main components provided in Table 5-2 and Table 5-3 include the specifications and typical chemistry of stainless steel, low alloy steel and carbon steel piping components. These piping components are processed to achieve a fine grain, microstructure.

**Reactor Pressure Vessel Weld Materials**

Weld materials used for the RPV construction as well as the RPV nozzle attachment processes are contained in Table 5-2 and Table 5-3. The ASME/ASTM specifications are listed that provide the details for the type of weld material, covered electrode, or filler metal, that are used in making the weld attachments.

**High Strength Reactor Pressure Vessel Bolting Materials and Fasteners**

The high strength low alloy bolting materials used in the RPV closure or for flange application are generally Type 4140 or Type 4340 high strength steels. The typical specifications and compositions are listed in Table 5-2 and Table 5-3.

The maximum measured yield strength and surface hardness of the stud bolting materials (studs, nuts, and washers) shall meet ASME Code requirements. Each stud is tested after rough machining on the outside surface and the center of an end position. Hardness tests is performed on all main vessel closure bolting to demonstrate that heat treatment has been properly performed. Given their importance, studs, nuts, and washers are individually tested to assure they meet specification limits.

### 5.2.4 Degradation Mechanisms for Reactor Pressure Vessel and Pressure Boundary Components

Component degradation mechanisms that are applicable to both the steel vessel and piping materials are assessed and controlled to assure component integrity. Degradation mechanisms include radiation embrittlement, general corrosion, and FAC. RPV $K_{IC}$ can be affected by the operating environment; therefore, $K_{IC}$ requirements are specified.

**Reactor Pressure Vessel Toughness Requirements**

ASME Section III requirements apply to all pressure boundary materials. Specific toughness values at the Reference Nil-ductility Transition Temperature ($RT_{NDT}$) are determined to assure the RPV maintains adequate fracture resistance at the lowest use temperatures. The different product forms have specific requirements that are evaluated, including the beltline shell courses, nozzle forgings, safe ends, and weld filler metal.

To assure that the steel components have adequate toughness, impurity limits are set on Cu, P and S for the base metal and Cu, P, V and S limits for the weld filler metal.

ASME Class 1 and Class 2 RCPB carbon steel components are made from high toughness grade material. In addition, ferritic components comply with ASME BPVC, Section III, requirements in accordance with the following:

- The ferritic materials used for piping, and valves of the RCPB are usually 64 mm (2.5 in.) or less in thickness but note that MS RIVs and CIVs may have a section thickness greater than 64 mm. Impact testing of ASME Class 1 components is performed in accordance with ASME BPVC, Section III, NB-2332 for thicknesses of 64 mm (2.5 in.) or less. Impact testing of ASME Class 1 components is performed in accordance with NB-2332 and NB-2331 for thickness greater than 64 mm (2.5 in.). Impact testing of ASME Class 2 components is performed in accordance with NCD-2330. In addition, ASME BPVC, Section III, Nonmandatory Appendix G, Paragraph G-3100 is to be met.
- Materials for ASME Class 1 and ASME Class 2 bolting with nominal diameters exceeding 25 mm (1 in.) are compliant with NB-2333 and NCD-2332.3, respectively, and are required to meet both the lateral expansion and the Charpy V values.
- The reactor vessel complies with the requirements of NB-2331 and 10 CFR 50, Appendix G. The reference temperature is established for required pressure-retaining materials used in the construction of Class 1 vessels. This includes plates, forgings, and weld material. The reference temperature differs from the nil ductility temperature in that, in addition to passing the drop weight test, three Charpy V-Notch specimens (transverse) must exhibit 68 J (50 ft-lbf) absorbed upper-shelf energy and 0.89 mm (0.035 in.) lateral expansion at 33°C (91°F) above the reference temperature. The core beltline material must meet 102 J (75 ft-lbf) absorbed upper-shelf energy. Consideration has been given to the effects of irradiation on beltline $K_{IC}$ by controlling the chemical composition residual elements of vessel beltline materials.
- Calibration of instruments and equipment meets the requirements of the ASME BPVC, Section III, NB/NCD-2360.

$K_{IC}$ tests will be performed for the BWRX-300 RCPB, and the test results are to be documented as required including description of the tests, locations of the test specimens and their orientation along with information on calibration of instruments and equipment.

**Radiation Embrittlement (Vessel Beltline) and Loss of Toughness**

For all material exposed to a neutron fluence of more than 1 x $10^{17}$ n/cm$^2$ for which impact tests are required, $K_{IC}$ shall be established as a function of temperature over the range of $RT_{NDT}$ to $RT_{NDT}$ + 33°C. Testing requires the use of specimens with adequate thickness with details given in ASME BPVC, Section II (Reference 5-18) and Section 5.11.8 entitled RPV Materials Surveillance Programs.

**General Corrosion**

The BWRX-300 addresses time dependent corrosion, other than localized corrosion degradation, such as IGSCC, Trans-Granular Stress Corrosion Cracking and Irradiation Assisted IASCC. For various operating conditions, corrosion allowances are incorporated into the design process. These allowances need to account for the impact of the Hydrogen Water Chemistry operating environment as well as the flow rates associated with standard operation (Hydrogen Water Chemistry might affect these rates). During design, general corrosion allowances for operation and shutdown are also considered. Alternate design allowances for specific components or systems are applied based on the local environment and operating conditions. To determine the total corrosion allowance, appropriate operating allowances for all plant conditions are added together to develop the overall required allowance.

**Flow Accelerated Corrosion**

General corrosion for the plant components is managed in the design process based on the ASME Code as stated in the previous section. However, under certain conditions of water chemistry, system flow, geometry, and component material, FAC can degrade piping and components made of carbon steel (with low chromium composition). The typical compositions for the carbon steel and alloy steel that are used for piping and piping components are given in Table 5-3. Options to ensure mitigation against FAC include increasing piping diameter, increasing elbow radius, using higher alloyed steels, or using stainless steel.

For piping made of carbon steel, extensive research has been performed to avoid FAC. The key factors that can affect the FAC rate are: (1) Temperature; (2) Material Composition; (3) Flow rate; (4) pH of the water; and (5) oxygen content. Each of these factors is discussed below:

- Temperature – generally in the range of 100ºC to 150ºC (can also occur at higher temperatures but becomes less significant)
- Material Composition – materials containing <0.25 wt percent of chromium
- Flow Rate – Fully developed turbulent flow (example: Reynolds number > 150,000)
- pH – environments with a pH <8
- Dissolved Oxygen – Concentrations < 30 ppb

These parameters provide the necessary understanding to allow the design team to assure that the concerns for any carbon steel lines are managed for the BWRX-300.

**Additional Time Sensitive Degradation**

Other than the risks of SCC, material data is required to address the effects of thermal conditions that could change the microstructure thereby reducing the ductility and toughness of the materials. This is applicable to the ferrite levels in cast stainless steel materials as controlled by the specifications U.S. Nuclear Regulatory Commission (USNRC) NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” (Reference 5-25). As discussed, FAC is a specific concern for carbon steels. Additionally, there are other types of loading that can affect material integrity. Specifically, cyclic loading, commonly referred to as fatigue, occurs in different regions and components in the BWR. These are addressed using ASME Code design rules.

**Bolting Relaxation**

Concerns for fastener relaxation are addressed by the initial material selection process, the design process that sets the applied stresses to assure elastic loading and the installation sequence and loading process. The design process also addresses any relaxation attributable to the operating environment. These guidelines are also complemented with the use of continual review of fastener performance in current operating BWRs.

### 5.2.5 Overview of Core Structural Component Materials

It is important that the materials to be used in the BWRX-300 are capable of maintaining reliable operation of plant systems and equipment during the design life of each component. For the reactor internals, austenitic stainless steel is primarily used to minimise levels of radiation from corrosion products generated in the core. Where higher strength requirements than those for standard austenitic stainless steels are needed, nickel base alloys, nitrogen strengthened austenitic alloys, or precipitation hardening stainless steels are used.

Materials selection with composition controls and defined fabrication processes is used to address materials degradation issues in the reactor system specifically SCC and irradiation SCC. Figure 5-5 shows the three factors that lead to degradation. Selection of the materials is directed toward minimizing their susceptibility. Design guidelines and controlled fabrication processes are used to manage the component stresses, with emphasis on surface stresses. It should be noted that the impact of the coolant environment is addressed using Hydrogen Water Chemistry/OLNC.

The stainless steel structural materials used in the BWRX-300 reactor core internals are primarily wrought materials along with their weld metals although Cast Austenitic Stainless Steels are used as well. It is noted that no Cast Austenitic Stainless Steels will be used for the RPV. However, Cast Austenitic Stainless Steels may be utilised in other pressure-retaining, cyclic, and/or dynamic load applications when supported by appropriate design analysis evaluations.

With respect to reactor internals and/or core support applications (see Section 5.2.5), Cast Austenitic Stainless Steels may be utilised. As an example, fuel supports are subject to some cyclic and dynamic loads during reactor operation and operating events. Cast Austenitic Stainless Steels is used for the fuel support due to the complexity of the part and challenges fabricating from wrought material. Appropriate chemistry controls to mitigate material degradation concerns will be applied. Operating experience with this material over the past 60 years supports its use in this application.

Table 5-4 lists the materials used for the typical internal components including the shroud, chimney, steam dryer, top guide, core support, CRD housings and instrumentation housings. The table lists the material product form as well as representative ASME/ASTM specifications.

Table 5-5 lists typical compositions for these austenitic reactor component materials.

**Wrought Austenitic Stainless Steels**

The primary alloys to be used for reactor internal components are austenitic structural alloys. Specifically, the wrought austenitic stainless steel components are Type 304, 304L, 316, 316L and potentially 316L with higher nitrogen content.

These materials have a long history of use in all past BWR applications. For L grade materials, the primary controlled constituent is carbon. Carbon is restricted to 0.02 weight percent to provide resistance to sensitisation during welding while still being able to meet the specified strength levels at the use temperature. For applications in high fluence regions of the core, residual elements are controlled to reduce the susceptibility of the steel. The efforts to maintain higher strength at temperature led to the development of special versions of Type 316L (originally designated Nuclear Grade which include nitrogen additions to maintain high temperature strength in these low carbon alloys). The chemistry of Type 316/316L material for components is also controlled to have positive ferrite potential as determined by ASTM A800/A800M-20, “Standard Practice for Estimating Ferrite Content of Stainless Steel Castings Containing Both Ferrite and Austenite,” (Reference 5-26) or the use of an approved alternate ferrite diagram for composition control with a target level of 3 percent. This is necessary to assure good weldability. The additional controls on alloy composition are also given in notes to Table 5-3. These austenitic materials are given a solution heat treatment at a temperature range with a hold time designed to achieve a full solution annealed condition while minimizing grain growth. The heat treatment is then followed by accelerated cooling to avoid any sensitisation. It should be noted that Alloy XM19 with its higher strength, developed for control rod drive components, will be used in special locations due to its excellent IGSCC resistance.

**Cast Austenitic Stainless Steel**

These cast versions of the wrought materials rely on carbon control and the presence of ferrite to assure IGSCC resistance which requires the ferrite level of 8 percent minimum for SCC. Additionally, the ferrite levels are also restricted to less than 20 percent to assure adequate toughness after long-term exposure to higher temperatures. The castings are given solution annealing to maintain IGSCC resistance. Prior operating experience has validated resistance to IGSCC (ASTM STP 756, “Intergranular Stress-Corrosion Cracking Resistance of Austenitic Stainless Steel Castings,” (Reference 5-27).

**Nickel Base Alloys**

Similar composition controls and product form processing to stainless steels are used to assure that wrought Ni-Cr-Fe Alloy 600 (including modified Alloy 600) and the associated Weld Metal Alloy 82 (ASME BPVC, “Case N-580-2 Use of Alloy 600 with Columbium Added Section III, Division 1,” (Reference 5-28) are free of any sensitisation risks. Nickel base wrought materials are Alloy 600 with niobium modified chemistry. Specifically, the materials need to have a stabilizing ratio (Nb + 2*Ti/C) > 92. Additionally, nickel base filler metals are to be Alloy 82 (ERNiCr-3) with a stabilizing ratio that also follows the same requirements.

**High Strength Alloys**

In special locations, higher strength Alloy X-750 is used for bolting and fastener type applications. The main approach used to prevent SCC susceptibility is accomplished by limiting the applied stresses in the design process. High strength austenitic Alloy X-750 is procured and processed to assure adequate SCC resistance. Additionally, Alloy 718 is considered to be used for high strength applications. Alloy 17-4PH is a martensitic precipitation hardening stainless steel of high strength, therefore, potentially susceptible to SCC if the hardness is not limited.

### 5.2.6 Overview of Design and Fabrication Process for SCC Controls in Reactor Core Components

To prevent SCC, the third factor that is necessary is the presence of elevated surface stresses. These residual stresses are often introduced by fabrication such as welding, surface grinding or forming operations. These types of stresses will support crack initiation on the wetted surface. For austenitic stainless steels or nickel base alloys, construction process controls are applied during various stages of component manufacturing and reactor construction to avoid these fabrication-induced surface stresses that could lead to SCC initiation.

These processing steps can also introduce localised sensitisation. The processes that need to be controlled include bending or forming, final machining, grinding, polishing, and welding. Each type of operation and an overview of the controls to be applied are summarised in more detail below.

**Forming Processes**

These processes are required for piping or cylindrical structures. It is important that forming is controlled and does not introduce significant strains that could lead to SCC initiation if the shape is not given a post forming solution anneal. These processes are limited to 2.5 percent total strain to prevent future SCC. The GEH materials and process controls define these limits.

**Machining, Grinding and Polishing**

Surface machining, and grinding are required as part of large component fabrication. Specific procedures employing specific media and subsequent polishing using increasingly finer media and decreasing material removal are also specified. These polishing procedures are particularly used following welding in the fabrication of reactor internal components such as the core shroud. Process qualification always precedes actual practice. The primary regions of potential susceptibility are the welds and their HAZ regions. Specifically, abrasive grinding of stainless steel HAZ regions is minimised to the extent possible. When grinding is applied to the weld regions, care is taken to confine grinding to the weld metal region only and limit grinding of the adjacent base metal. This is necessary to meet the required fabrication and examination requirements of ASME. Grinding abrasives are controlled to avoid detrimental effects on austenitic materials. Grinding is followed by polishing processes to further reduce the risks of SCC.

The use of the final polishing processes is restricted to those with qualified procedures. The procedure contains methods that involve finishing the ground surface with successively finer grit size such that the bulk of any surface cold work is removed. Finishing with each grit size continues until evidence of the previous grit is removed.

**Welding Processes**

Over the history of BWR plant construction and operation, GEH has refined the technologies for fabrication, maintenance, and repair of welded components in nuclear reactors. Each process improvement has increased margins against SCC in BWRs by controlling residual stresses, alloy chemistry, and microstructure for the welds and surrounding HAZ regions. GEH developed specialised narrow groove welding technology, which is an automated welding technology addressing these three critical factors. Specifically, it produces compressive stresses in the weld root regions. The narrow weld joint design improves fabrication productivity and reduces weld shrinkage distortion. The low heat input, combined with special welding parameters, also limits HAZ residual strains and sensitisation in addition to the compressive stresses, which will further prevent SCC. Since its development, the process has been used across the power generation industry.

**Electro-Discharge Machining**

Under special circumstances, Electro-Discharge Machining is used. Specific controls, developed in previous BWR applications, are used on nickel base alloys to prevent surface degradation such as microfissuring.

In summary, GEH uses developed fabrication processes with a long history of refinement to prevent SCC initiation in components made using austenitic structural alloys. These fabrication processes are also used to prevent IASCC and Trans Granular SCC.

**Additional IASCC Material Controls**

As with SCC, the parameters that contribute to IASCC in a BWR environment are material selection with its processing, the applied and residual stresses in the components, and the magnitude of the neutron radiation exposure. Type 316L is the alloy of choice for the core region. Table 5-5 includes a listing of special chemistry controls on silicon for Type 316L when used for reactor internals subjected to IASCC. The benefits of the control of silicon are confirmed by research and are used in core components, specifically the core shroud.

**Specific Trans Granular Stress Corrosion Cracking Fabrication Controls**

IGSCC and Trans Granular SCC often occur in the same alloys depending on the environment or the microstructure. However, cases of Trans Granular SCC in small lines and other components, have been associated with contaminants such as chlorides that can be present on the material surface.

These contaminants, related to salts or marking materials, are also controlled during fabrication as well as operation. Miscellaneous non-metallic materials, used for the fabrication and assembly of reactor system components, are also controlled to minimise contamination with species known to have detrimental effects on alloys such as stainless steels and nickel alloys.

Those materials which come in contact with stainless steel or nickel alloys during these processes are either low in halogens, sulfur, and lead or other low melting alloys or need to be removed in the final cleaning of the part or assembly. The known contaminants of concern are chlorine, fluorine, sulfur, lead, mercury, and all other low melting point metals. In addition, when welding or solution heat treatment is involved, phosphates and carbonaceous materials are to be removed prior to welding or heat treatment.

**Sensitivity to Fracture**

The alloys used in BWR systems other than the RPV and steel components are predominantly alloys with lower strength and high ductility. These alloys are not susceptible to rapid fracture as substantiated by the excellent operating history of BWR components fabricated from austenitic stainless steel and nickel base alloys. The only components that show loss of ductility and reduced fracture resistance are the core components, particularly the core shroud, which is subjected to irradiation. The irradiation leads to material changes, hardening and grain boundary segregation, that in, turn, increase the strength significantly and reduces the ductility. The same factors that increase the risk of IASCC, also reduces fracture resistance. The shroud design takes these factors into account to assure adequate robustness. The design also positions the shroud welds to reduce the fluence experienced by the weld region. Cast Austenitic Stainless Steels material compositions restrict the ferrite level to assure toughness is not lost with thermal aging.

### 5.2.7 Gasket, Seal, and Fastener Materials

**Gaskets and Seals**

These non-metallic engineered materials used in association with reactor system components are controlled to minimise or eliminate potential for detrimental effects on metallic reactor components. These materials which include gaskets, packing, and bushings that are installed within the reactor system such that they are in contact with reactor water conform to the chemistry controls of specifications. Examples of the specific materials used include Viton, Buna-N, EPDM, and graphite non-metallic seals as well as metallic seals including silver plated Alloy 718. The specifications set limits on halogens, sulfur, and nitrates in the non-metallic materials. There are also specific prohibitions or limitations on the use of chlorine and fluorine bearing materials such as Teflon and polyvinyl chloride. Exceptions are allowed but must be evaluated on an individual basis considering such factors as location, radiation dose, temperature, and successful operating history.

**Fasteners**

Sections 5.2.3 and 5.2.5 include discussion of low alloy steel and austenitic high strength materials, respectively, that are used as fasteners. There is extensive experience in operating BWRs with these materials that is used as the basis for fastener selection for use in the BWRX-300. Table 5-2 through Table 5-5 include the specifications and compositions of these special use materials.

## 5.3 Reactor Coolant System and Reactor Coolant Pressure Boundary

The NBS is comprised of three primary subsystems: RPV, MS, and RPV instrumentation. In addition to these subsystems, the NBS also provides RPV head venting and RPV head flange seal leak detection features. Within the NBS are the redundant with diverse instrument and electrical features, SC RIVs interfacing with the subsystem and systems listed below:

- MS (Section 5.1.3 and PSR Ch. 10, Section 10.1)
- RPV head vent (Section 5.12.4)
- ICS steam supply (Section 5.7 and PSR Ch. 6, Section 6.2.1)
- ICS condensate return (Section 5.7, PSR Ch. 6 and PSR Ch. 9)
- CFS (PSR Ch. 10, Section 10.2.2)
- CUW (Section 5.12.2 and PSR Ch. 9A, Section 9A.2.2).

The NBS descriptions presented within this section are consistent with the content of 006N7828 (Reference 5-8).

The RCPB components are discussed in section 5.1.1.

### 5.3.1 Nuclear Boiler System Configuration

The NBS (see Figure 5-2) consists of the RPV, two MSLs, MSRIVs, Feedwater Reactor Isolation Valves (FWRIVs), ICSRIV, ICRRIV, CUW RIVs, MSCIVs, a steam line drain/bypass subsystem, RPV Head Vent RIVs and Head Vent subsystem, a RPV flange leak detection subsystem, nuclear instrumentation, and reactor pressure, reactor water level, steam and core flow, temperature, and leak detection instrumentation. Several pressure and temperature measurements are utilised to calculate the heat balance, which determines the RPV core flow.

Material and equipment selection for the system components is based on a 60-year design life, with appropriate provisions for maintenance and replacement.

**Reactor Pressure Vessel Description**

The BWRX-300 RPV (Figure 5-6) assembly consists of the RPV with nozzles, RIVs and other appurtenances, a removable closure head, the reactor internals, and supports. The RPV instrumentation that monitors the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for the FMCRDs.

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

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

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

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

The major reactor internal components include the following.

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

The NBS utilises a RPV to house and support the core and internals, provide a pressure boundary and flow path for reactor coolant, interface with FW as the water supply, generate steam for the main turbine, and provide emergency core cooling via the ICS.

The underside of the RPV closure head is not clad. The closure head is flanged and bolted to the shell for removal purposes. The head contains a blind flanged spare nozzle to be used for future instrumentation purposes when required.

The RPV provides a flange leak detection connection to communicate to a groove between the two concentric O-ring seals so any leakage from the inner seal may be sensed as a buildup of pressure with the outer seal. The head closure is designed such that leakage is prevented from either seal with the design being capable of withstanding full reactor pressure out to the outer seal at all times. Seal effectiveness is determined by a closure head seal leak detection subsystem, which monitors for leakage between the O-rings.

The RPV bottom head contains penetrations for the CRDs and the nuclear in-core instrumentation.

**Main Steam Subsystem**

The MS subsystem is comprised of two MS lines that are routed from the outboard MSRIV discharge through the outboard MSCIV out to the TSVs, associated branch piping, and the MSL drains.

The MS piping (in Containment and RB) from the discharge of the outboard MSRIV to the SIR is designed to ASME BPVC, Section III, Division 1, Class 2, and Seismic Category 1A requirements following USNRC NUREG-0800 BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," (Reference 5-29) guidance.

The MS piping from the outboard MSRIV discharge to the low point of the piping just inside the TB is designed to have a negative slope such that any steam condensate will drain outwards to the MS drain lines.

The MS piping in the TB, including all the connecting branch lines up to the various steam loads and their respective isolation valves (see PSR Ch. 10, Section 10.1.3) is designed to ASME B31.1 (Reference 5-9), which has non-seismic category requirements. The MS equalizing header supplies high-pressure steam via branch piping to the TBVs, the Moisture Separator Reheater System (MSR), No. 6 Feedwater Heater (FWH), and the Turbine Auxiliary Steam Subsystem (TASS).

The MS pipeline routing includes sufficient length such that the MSL minimum volume requirements are met using the pipe sizes designated from the system Piping and Instrumentation Diagram and Process Flow Diagram.

The MS subsystem is designed so that the pressure drop through the piping at 100 percent Rated Thermal Power (RTP) condition does not exceed the minimum or maximum limits.

MSLs are designed to minimize the potential for water (steam) hammer through the implementation of specific design features and system layout(s). The design also complies with USNRC NUREG–0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," (Reference 5-30) as applicable.

The BWRX-300 design includes an outboard MSCIV (see Figure 5-2) on each MSL. The MSCIVs provide Safety Category 1 containment isolation for accidents or events to prevent unfiltered release of radiological contaminants through the MSLs that exceed limits. This containment isolation is single failure proof. The MSCIVs are fast-closing and fail-closed type valves. These isolation valves outside of containment are located as close to the containment as practical to satisfy the containment isolation design requirements.

The MSCIVs are designed in accordance with the rules and requirements of ASME BPVC-III BN, (Reference 5-7), in accordance with their quality group classification. The MSCIVs are designed for Seismic Category 1B and certified by a qualification program prepared and performed in accordance with ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," (Reference 5-31).

The MS drain lines provide the ability to drain the condensate from the MS lines to the condenser in a controlled manner during startup, lower power operation (approximately 40 percent RTP for drain valves control), normal power operation, and shutdown. The MSL drain isolation valves incorporate control logic interlocked to support automatic plant startup and shutdown operation.

**Reactor Coolant System Instrumentation Subsystem**

*Reactor Water Level*

The water level in the reactor vessel is measured by the temperature compensated dP devices (calibrated for specific RPV pressure and temperature conditions).

Historically, BWR water level has been monitored over both NR and WR. Previous RPV designs have multiple taps to cover the different ranges. The NR is calibrated for a smaller range of elevations and therefore can be held to a higher accuracy than WR. This is because the measurement accuracy is linearly related to the overall differential height of the sensor pair. During normal operations, NR is used to control the target water level and FW flow. WR level monitoring is used during off-normal events. In the case of BWRX-300, the added safety features of all RPV shell nozzle penetrations located above an elevation of TAF + 4m and minimizing the number of water level instrument nozzles results in having only one pair of nozzles for each train of water level monitoring. Therefore, NR and WR for each division are taken from the same set of nozzles, with the NR pressure sensor pairs calibrated over a smaller range.

The reactor water level measuring system is classified SC1 and is monitored by the DCIS. The DCIS is also used to monitor the water level for both Safety Category 2 and Safety Category 3 functions. The BWRX-300 is designed to have three divisions for monitoring and control of Safety Category 1 functions, as well as redundancy within each division (refer to PSR Ch. 7 for detailed description of control logic architecture).

Another key function of the water level measurement system is to provide shutdown range water level monitoring as the RPV is flooded up before the head is removed. It will also monitor water level during refueling and support post-outage restart. This is a Safety Category 3 function with two pressure transducer pairs.

Water level measurement is not required between TAF and TAF + 4m because there are no operator actions associated with this range. In the case of a Beyond Design Basis Accident and DECs where monitoring water level in the core (also known as fuel zone range) are necessary, it will be accomplished using an appropriate proven technology such as Heated Junction Thermocouples.

*Reactor Pressure Instrumentation*

RPV steam dome pressure is measured using Wide Range (WR) and Narrow Range (NR) dP instruments to provide pressure indication in Modes 1 through 5. The RPV steam dome pressure measurement supports Post-Accident Monitoring system.

*Core Flow Instrumentation*

Mass core flow in the BWRX-300 RPV is not measured directly but is calculated from an aggregation of temperature measurements throughout the core and RPV pressure. The core flow is calculated by the heat balance core flow methodology using the core inlet temperature measurement as input to determine core inlet enthalpy. The methodology provides a verified basis for calculated core flow.

*Main Steam Flow Instrumentation*

The MS flow restrictors are located in the RPV MS nozzles. They perform functions in various safety categories and corresponding defence levels. The Safety Category 1 functions are to limit the overall steam flow through one steam line when the steam flow exceeds preselected operational limits (i.e., if a large break of a MSL is sensed). The other Safety Category 1 function is to provide that same sensing function to the SC1 MSRIV/MSCIV isolation system. The Safety Category 2 steam flow function also supplies the same sensing functions for MSRIV/MSCIV isolation in addition to the already discussed Safety Category 1 isolation function. The Safety Category 3 steam flow function supports the overall reactor water level controller that maintains target water level as one of the inputs to that controller.

The MS flow restrictor provides low-pressure sensing taps for instrumentation lines of dP transmitters that are used to measure steam flow. MS flow is measured through the RPV nozzle flow restrictors using Bernoulli’s equation of dP to derive the MS mass flow. The mass flow can then be calculated by knowing the steam properties based on RPV pressure and temperature.

*Reactor Power Instrumentation*

Reactor power instrumentation is described in Section 5.4.2.

*RPV Metal Temperature Instrumentation*

The RPV outside surface (metal) temperatures are measured at the head flange and the bottom head locations in order to comply with plant Operational Limits and Conditions (OLCs) (Technical Specification) surveillance RPV heat-up and cooldown requirement limits which define allowable operating regions and permits a large number of operating cycles while also providing a wide margin to nonductile failure.

### 5.3.2 NBS Classification of SSC

The overall NBS is classified as a SC1 system following the methodology for classification of BWRX-300 SSCs as discussed in PSR Ch. 3, Section 3.2.

### 5.3.3 NBS Functions

The NBS components are designed, specified, procured, fabricated, tested, installed, and commissioned in accordance with the codes and standards specified in 006N3441, "BWRX-300 Applicable Codes, Standards, and Regulations List," (Reference 5-32). 006N3441 (Reference 5-32) lists the approved codes and standards and identifies the applicable editions used for the design of mechanical systems and components of the BWRX-300 including those described in this chapter. The pressure boundary codes and standards are listed in Table 3-10 of Attachment 1 to PSR Ch. 3.

**Normal Functions**

The NBS performs the following Normal Functions in normal operating modes:

- Provides the NBS portion of the RCPB
- Provides RPV internal structures
- Directs the natural circulation of reactor coolant through the core
- Measures reactor pressure, reactor level, and reactor power
- Provides the means to convey steam from the reactor to the main turbine
- Prevents the accumulation of non-condensable gases within the system
- Provides RPV access necessary to support outage activities
- Provides monitoring of parameters.

The normal operating modes including description of how the plant operates in these different operating modes is provided in Section 16.6 of NEDC-34188P, "BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions," (Reference 5-33).

**Off-Normal Functions**

The RCS performs DL3 Safety Category 1 functions that support high availability of the BWRX-300 Fundamental Safety Functions as listed below:

- Control of reactivity
- Fuel Cooling
- Long-term heat removal
- Containment of radioactive materials (isolation of RCPB)

As described in PSR Ch. 3, Section 3.2, the allocation of Defence Line (DL) functions to system function categorisation and system classification is as follows:

- SC1 equipment is located in DL3
- SC2 equipment is located in DL4a
- SC3 equipment is located in DL2

### 5.3.4 System Operations

**Initial Configuration (Pre-Startup)**

System configuration is established per plant procedures. Operating modes are defined in PSR Ch. 16, Section 16.6.

**System Startup**

Startup of the NBS begins with reactor operations in Mode 2. System startup consists of reactor startup from cold shutdown (RPV coolant bulk temperature less than or equal to 90.0°C) or hot shutdown conditions (bulk temperature greater than 215.6°C). All components and instruments of the NBS are operable and valves are aligned to the positions (open or closed) required.

During startup operation, the NBS is heated to 90°C and condenser vacuum established so the RPV can be deaerated before nuclear heating is started. At this time the MSRIVs and MSCIVs are opened. Reactor startup is begun by withdrawing control rods in a predetermined sequence to achieve core criticality and continued to the onset of coolant boiling. The heat-up rate after boiling onset is automatically controlled using the TBVs and condenser. The reactor water level is controlled by manual operator or automatic direction to maintain the level in the normal range during the transition from SDC to FW injection. Continued control rod withdrawal raises core thermal power up to and not exceeding 12% RTP.

RPV level control is performed by regulating the FW flow. With the reactor core power stable at about 12 percent RTP, preparations are made for turbine roll and generator synchronisation. The MSL drain line valves and the isolation valves to other system interfaces are open. The MSL drains remain open to approximately 40% RTP to remove excess moisture carryover.

**Normal Operations**

Power Operation (Mode 1) consists of normal operation, planned transients, AOOs and infrequent events (unplanned transients), and on-line testing. Normal operation includes steady-state operation at a given power level up to RTP. All required components and instruments of the NBS are operable and are aligned as required for system operation. Subcooled FW is provided by the CFS system to keep the RPV level in the normal operating range. The reactor pressure is controlled at 7.17 MPa with a constant steam mass flow rate.

As the water boils in the core, the chimney provides an area of expansion such that the velocity lowers and allowing the void fraction to increase. The NBS also provides the means for separating the steam from the liquid and drying the steam by channeling flow through the steam separator and steam dryer before it goes to the MS lines and the turbine. The MS subsystem transports steam generated in the RPV through PCS discharging from the RPV nozzles through the MSRIVs, traveling through the MSLs and MSCIVs to provide the steam to drive the main turbine. Steam is directed through the open maintenance block valves to the MSR, number 6 FWH, and TASS. MS line drain line valves open and close as a function of the level of condensed steam in the drain pot whilst the plant is below approximately 40% RTP. The MS line drain valves are closed or verified closed when the plant is at or above approximately 40% RTP.

During power operation, the non-condensable gases generated in the RPV will be vented through the RPV head to the MS piping.

**Off-Normal Operations**

*Planned Transients*

Planned transients include reduction or increases in power, rod pattern changes, reduction to 0 percent power and hot standby.

Reductions in power to 50 percent by rod pattern changes have no effect on NBS operation.

Reduction in power to 40 percent and below results in automatic opening of the drain orificed line valves to the main condenser.

The TBVs open whenever the TSVs and Turbine Control Valves (TCVs) close, bypassing steam to the condenser.

*Unplanned Transients*

Unplanned transients are listed in PSR Ch. 15.5.

**System Shutdown**

System Shutdown contains three (3) states:

- Hot Shutdown
- Stable Shutdown
- Cold Shutdown

*Hot Shutdown*

During Hot Shutdown (Mode 3), the NBS is intact and available to be pressurised.

Reactor shutdown from the power operation begins with the reactor at rated pressure and temperature and the reactor Mode Switch in RUN position. Rod insertions reduce plant power until the turbine is taken offline. RPV coolant level is normally controlled using FW to provide reactor water level control if the main condenser is available to reject core decay heat. CUW can be used to overboard excess coolant level if the RPV is pressurised.

Insertion of control rods is performed to establish a reactor cooldown rate and slowly depressurise the reactor. After all control rods are inserted, the reactor is placed in the Hot Shutdown position. If conditions require operator action to expedite core shutdown, the control rods are rapidly inserted by a trip signal initiated by placing the reactor Mode Switch to the SHUTDOWN position.

In Hot Shutdown, the RPV coolant bulk average temperature is less than rated conditions but greater than 215.6°C. The NBS continues to provide steam paths for the removal of core decay heat by the main condenser or by the ICS, depending on the status of the NBS isolation subsystems. Makeup is provided as required either from the CFS or from CRD system.

*Stable Shutdown*

Entering Stable Shutdown (Mode 4), starts with the reactor exiting Hot Shutdown usually before the reactor Mode Switch is placed in the SHUTDOWN position. The NBS is intact and able to be pressurised. Insertion of control rods is continued to establish a reactor cooldown rate and slowly depressurise the reactor. If FW is available, decay heat can be removed using the TBVs to discharge steam to the condenser and return back using FW and/or condensate. MS may or may not be isolated depending on the availability of offsite power and whether the condenser is available for decay heat removal. If offsite power is not available, decay heat removal is performed by the ICS.

*Cold Shutdown*

Normally, the reactor design is to be cooled down from the HOT SHUTDOWN condition by opening one or more TBVs to direct steam to the main condenser. RPV cooldown is carefully controlled so that it is less than the design limit of 111.1°C/hr. SDC initially provides a small assist to this cooling. When the reactor pressure reaches approximately the pressure at which turbine gland sealing steam is no longer effective at maintaining vacuum in the main condenser, the remaining residual heat in the reactor is within the capability of SDC. Both FW lines provide a path for SDC flow to return to the RPV. Reactor cooldown continues with SDC until entering Cold Shutdown (Mode 5) with reactor bulk coolant temperature at or below 93.3ºC. Coolant temperature is maintained above the design minimum temperature with the head bolts tensioned. The NBS instrumentation monitors the reactor water level, temperature, and pressure.

**Refueling Mode Operation**

When the RPV is to be filled for refueling, the RPV is further cooled to near temperature inside containment using SDC. When the reactor vessel head stud de-tensioning begins, the plant operating state is Refueling Mode (Mode 6). The RCS instrumentation is used to monitor the reactor water level and temperature. No further RCS action is needed unless RIV maintenance is planned in which case nozzle plugs are installed.

### 5.3.5 Testing and Maintenance

This section provides, in part, the basis for preparation of detailed maintenance procedures. It defines the maintenance philosophy, outlines the procedures for scheduled (preventative) and unscheduled (corrective) maintenance, and Inservice Inspection (ISI) and surveillance. It also identifies interfacing systems needed to support maintenance operations.

The BWRX-300 ISI requirements specification provides mandatory Pre-Service Inspection, ISI, and Inservice Testing (IST) requirements for components and systems. Pre-Service Inspection, ISI, and IST requirements are consistent with ASME BPVC-XI-1 (Reference 5-21) and ASME OM, “Operation and Maintenance of Nuclear Power Plants,” (Reference 5-34).

The Pre-Service Inspection, ISI, and IST requirements include examinations, inspections, and testing of the RPV and reactor coolant transporting systems, components, piping, and pipe supports, which are designed and installed in accordance with ASME BPVC, Section III, and applicable codes and standards listed in 006N3441 (Reference 5-32). In addition, this specification includes basic design control considerations within ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications Part 2,” (Reference 5-35). The ISI requirements ensure plant nuclear safety, minimal radiation exposure to the general public, and plant personnel, and proactive management of potential adverse plant system deterioration and aging effects.

Any required surveillance testing is determined as part of the work planning / work release approval process. Post-Maintenance Testing is determined as part of the maintenance review and approval process.

Pre-service and ISI are performed in accordance with ASME BPVC, Section XI.

The RIVs have appropriate access platforms and load handling equipment to assist valve assembly maintenance and load transport inside the containment.

The RCS water quality meets the requirements delineated in the BWRX-300 Water Quality Design Specification.

Areas requiring ISI inspection are provided with access spaces/routes and with removable insulation coverings.

### 5.3.6 Compliance with Regulations

The BWRX-300 RPV meets the requirements of US 10 CFR50 Appendix A General Design Criteria (GDC) and applicable USNRC Regulatory Guides.

The RCPB as depicted in Figure 5-4 is comprised of ASME Class 1 and ASME Class 2 components as detailed below.

The following RCPB components are Quality Group A, ASME Class 1 components designed to meet the requirements of ASME BPVC, Section III, Division 1, Subsection NB:

- The RPV
- The RIVs
- The piping from the outboard RIVs up to and including each respective outboard CIV (this includes the redundant isolation valve downstream of the CIV connected to the ICS to SDC interface piping).
- The ICS including steam supply lines, ICs, purge lines, purge valves, condensate return lines, condensate return valves, ICS to SDC interface piping (including CIV and redundant isolation valve), ICS to BIS interface piping (including CIV)

- The CRD interface flanges/housings
- The nuclear instrumentation interface flanges/housings

The following RCPB components are Quality Group B, ASME Class 2 components designed to meet the requirements of ASME BPVC, Section III, Division 1, Subsection NCD and RG 1.26:

- The piping from the outboard RIVs up to each respective CIV (except for ICS components)
- The outboard CIVs for MS, CUW, and CFS
- The head vent piping to the MSL and head vent interface piping to quench tank up to and including the isolation valves for the quench tank
- The reactor instrumentation lines

The RCPB piping outboard of the RIVs, and SIR are classified ASME Class 2, excluding ICS piping up to the ICS. The RPV level instrumentation sensing lines are classified as ASME Class 2 per ASME BPVC, Section III, NB-3630(d).

### 5.4 Reactor Vessel

The BWRX-300 RPV (Figure 5-6) assembly consists of the vessel with RIVs and its other appurtenances, a removable head, the reactor internals and supports and instrumentation. The RPV instrumentation that monitors the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for the FMCRDs.

The RPV is a vertical, cylindrical pressure vessel fabricated from forged rings or rolled and welded plate rings joined together, with a removable top head by use of a head flange, seals, and bolting. The RPV also includes penetrations, nozzles including RIVs, and reactor internals support. The reactor vessel is relatively tall which permits natural circulation driving forces to produce core coolant flow.

The RPV is designed to keep its structural integrity during all specified loading conditions. It also maintains its integrity against nonductile failure including consideration of material embrittlement due to neutron irradiation during its design life.

A longer internal recirculation flow path (relative to forced-circulation BWRs) is provided by a “chimney” in the space that extends from the top of the core to the entrance of the steam separator assembly. The top guide, the chimney, chimney head and steam separator assembly are supported by a shroud assembly that extends to the bottom of the core.

#### 5.4.1 Major Reactor Internal Components

This section addresses the RPV internals which consist of all the structural and mechanical elements inside the RPV.

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and in-core nuclear instrumentation) are as follows:

- Core support structures (shroud support, core plate, shroud, top guide, chimney, control rod guide tubes, control rod drive housing, orificed fuel supports and peripheral fuel supports)
- Internal structures (chimney head and steam separator assembly, steam dryer assembly, FW spargers, internal piping, nuclear instrumentation, in-core guide tube stabilizers, surveillance sample holders)

A simplified representation of the BWRX-300 RPV and internal component locations is shown in Figure 5-6.

#### 5.4.2 Core Components (Control Rods and Nuclear Instrumentation)

Major Core Components including, Fuel Bundles and Control Rods are discussed in PSR Ch. 4. The control rods (see PSR Ch. 4.5, Figure 4-6) insert negative reactivity into the core to control reactor power.

The nuclear (in-core) instrumentation consists of the PRNM, Gamma Thermometers, and WRNM. The Local Power Range Monitors (LPRMs) connect to the SC1 I&C System as DL3/SC1 DCIS instruments. The Gamma Thermometers are SC2 and are used to calibrate the LPRMs as well as to support DL4a/SC2 diverse scram functions. The WRNMs are DL2/SC3 instruments and provide monitoring up to 20% power during startup. The LPRMs provide reactor power data used by the SC1 I&C system to perform DL3 functions. Additional information can be found in 006N5114, “BWRX-300 Plant I&C Systems Architecture Requirements and Design,” (Reference 5-36).

The WRNM system monitors the 10 fixed neutron monitoring detectors in the core. The detectors are distributed radially in the core at fixed heights with at least two detectors in each quadrant (eight total), with two additional detectors at the boundaries of two adjacent quadrants such that each quadrant has a backup in the event of a single WRNM failure.

### 5.4.3 Core Support Structures

The Core Support Structures (shown in Figure 5-6) include:

- Shroud
- Shroud Support
- Top Guide
- Core Plate
- Control Rod Guide Tube
- Orificed/Peripheral Fuel Supports
- Chimney Lower Flange
- CRD Housing.

The design life of core support structure components is no less than 60 years.

**Shroud**

The shroud is the cylindrical support of the reactor core area. It is supported by the shroud support assembly. The core plate is between the shroud and the shroud support, and these three pieces are bolted together. The top guide is between the shroud and the lower chimney flange, and these three pieces are bolted together. The shroud provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. The shroud supports the weight of the chimney head and steam separators, chimney, and top guide.

**Shroud Support**

The shroud support is a circular assembly with vertical stanchions (legs) that are welded to the reactor bottom head. The shroud support provides the vertical and lateral support for the shroud and other components like the top guide and core plate. It also supports the chimney and chimney head and steam separator assembly, and steam dryer assembly.

**Top Guide**

The top guide is at the top of the reactor core. It provides the lateral support for the fuel assemblies, nuclear instrumentation, startup sources, and control rods. The top guide is attached between the top of the shroud and the chimney lower flange.

**Core Plate**

The core plate is at the bottom of the reactor core area. The core plate consists of a circular plate with round openings. The core plate is bolted between the support structure's ring and shroud and also forms a partition that causes the core flow to pass into the orificed fuel supports and through the fuel assemblies. It provides the lateral support for the fuel assemblies and control rods through the orificed fuel supports, provides vertical and lateral support for the nuclear instrumentation, and provides vertical and lateral support the startup sources, and for the peripheral fuel supports and fuel bundles.

**Control Rod Guide Tube**

The control rod guide tube fits in holes in the core plate and rests on the control rod housings that are welded to the reactor bottom head. They provide the lateral support and channel water to the control rods as they move up and down within the core to change the reactor power levels. The control rod guide tube also provides vertical support to the orifice fuel support and to four fuel bundles landed on the orifice fuel support. Each guide tube is designed as the guide for the lower end of the control rod and as a support for an orificed fuel support. The control rod guide tube contains holes near the top of the guide tube but below the core plate to direct coolant flow to the orificed fuel supports.

**Orificed/Peripheral Fuel Support**

Orificed fuel support rests inside the respective control rod guide tube with a flanged top end that sets down onto the control rod guide tube top end to transfer the vertical load. They provide a spot for the lower end of each fuel assembly to rest and also provide cruciform slots to guide the control rod blade as it rises or lowers to increase or decrease reactor power. The orificed fuel support slots are crucial to keeping the control blades properly aligned between the four bundles for rapid insertion into the core from fully withdrawn.

The peripheral fuel supports rest in the core plate around the outside of the orificed fuel supports and fuel bundles. A single fuel bundle rests in a peripheral fuel support instead of the four fuel bundles that rest in each orificed fuel supports. The peripheral fuel support contains a flow restriction designed to ensure proper coolant flow to the peripheral fuel assembly. Otherwise, their function is the same as an orificed fuel support.

**Chimney Lower Flange**

The chimney lower flange is welded to the bottom of the chimney barrel and is mounted on the top guide and bolted to the top of the shroud. The chimney lower flange provides the vertical and horizontal support for the chimney barrel.

**Control Rod Drive Housing**

Provides the extension of the RPV pressure boundary at the bottom head for installation of its respective CRD. They also support the weight of a control rod, CRD, control rod guide tube, thermal sleeve, orificed fuel support, and four fuel assemblies which is transmitted to the bottom head of the vessel. Finally, they provide the ports or the attachment of CRD supply hydraulic lines.

### 5.4.4 Chimney Barrel and Chimney Upper Flange

The SC1 chimney (Figure 5-6) is a long cylinder mounted to the top guide that supports the steam separator assembly. The chimney forms the annulus separating the subcooled recirculation downward flow from the upward steam-water mixture flow exiting the core. The recirculation flow consists of reactor coolant returning from the steam separators and FW makeup. The cylinder is flanged at the bottom and top for attachment to the top guide and the chimney head, respectively.

The chimney barrel is an extended height cylindrical component welded to the lower chimney flange. The chimney barrel forms the annulus separating the subcooled recirculation flow returning downward from the upward steam-water mixture exiting the core. It provides the necessary additional flow space (volume) within the RPV that permits natural circulation driving forces to produce abundant core coolant flow. The chimney upper flange is welded to the top of the chimney barrel and supports the chimney head and separator assembly.

### 5.4.5 Chimney Head and Steam Separator Assembly

The chimney head and steam separator assembly is a perforated domed plate including a flange that is bolted to the chimney upper flange provides a path of the steam flowing from the core to the steam dryer assembly. The standpipe and separator assemblies are supported on and attached to the chimney head. The steam separator part of the assembly has no moving parts and uses centrifugal force to decrease most of the moisture in the steam before it enters the steam dryer assembly. The separated water drains to the down-comer annulus.

#### 5.4.6 Steam Dryer Assembly

The steam dryer assembly completes the function of removing the moisture from the steam before it exits the reactor. The moisture content is lower than 0.1% weight percent at 100% reactor power when the steam leaves the reactor on its way to the main high-pressure turbine. The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure that is removable from the RPV as an integral unit. The assembly includes the dryer banks, drain collecting troughs, drain ducts and a skirt that forms a water seal extending below the separator reference zero elevation. Upward and radial movements of the dryer assembly under the action of dynamic loads is limited by reactor vessel internal stops and brackets that are arranged to permit differential expansion growth of the dryer assembly with respect to the RPV.

#### 5.4.7 Feedwater Spargers

The FW spargers deliver makeup water to the reactor during plant startup, power generation and plant shutdown modes of operation. The FW spargers also provide uniform distribution of FW flow within the downcomer flow passage and deliver CUW makeup flow and shutdown cooling flow to the vessel.

#### 5.4.8 Head Vent Internal Piping

The internal vent piping inside the RPV closure head is a discharge line that is integrated into and through the bracket of one of the internals guide rods. The internal piping is, thus, split into a portion mounted inside the RPV closure head and a portion mounted below the vessel flange with an intersection of the internal piping at a guide rod bracket where the two ends are joined. The internal vent piping is classified as SC3, and Seismic Category 2.

The piping in the RPV closure head is mounted with at least two brackets that hold the vent in place while allowing for thermal expansion and mitigating flow induced vibration effects. The brackets are designed to the rules ASME BPVC, “Section III, Division 1, Subsection NF, Rules for Construction of Nuclear Facility Components,” – Division 1 – Subsection NF – Supports (621) (Reference 5-37), classified as SC1, and Seismic Category 1A.

#### 5.4.9 Reactor Water Cleanup System Suction Piping

Two CUW internal drain lines are routed from flange connection inside the vessel to each respective nozzle, through an inverted loop, down the vessel shell internal wall, and to the RPV bottom head.

#### 5.4.10 Nuclear Instrumentation Tubes and Stabilizers

The in-core guide tubes extend from the top of the in-core housing to the top of the core plate. They provide the in-core instrumentation with protection from flow of water in the bottom head plenum, and guidance for insertion and withdrawal from the core. The in-core guide tube stabilizers provide lateral support and rigidity to the in-core guide tubes.

#### 5.4.11 Surveillance Sample Holders

The surveillance sample holder is a non-power generation internal device whose purpose is to enclose vessel material specimens. The specimens are irradiated during the life of the plant. Their location permits sample removal for examination incrementally over the life of the vessel. The intent of the surveillance program is to monitor and evaluate radiation-induced changes in the vessel material. The material surveillance sample holders are located next to the RPV inner wall around the periphery of the core at the core midplane.

## 5.5 Reactor Coolant Circulation

Reactor coolant flow through the BWRX-300 core is by natural circulation, meaning pumps (reactor coolant or recirculation) are not required to force reactor coolant through the core. Natural circulation is enabled mainly by the addition of a high chimney between the top of the core at the top plate to the bottom of the steam separator.

## 5.6 Reactor Coolant Piping

The BWRX-300 RCPB and associated piping is described in Section 5.1.1, Section 5.3 and PSR Ch. 6. Following insertion of control rods, the BWRX-300 cools and depressurises the reactor by natural circulation with decay heat removal by the ICS or SDC. During overpressure events or loss of normal heat sink events, the ICS is used to quickly depressurise the RCPB and cool the core.

## 5.7 Reactor Pressure Control System

### 5.7.1 System and Equipment Functions

This section describes systems that protect the RCPB from over-pressurisation. SC3 I&C control system regulates RPV pressure by controlling the position of the TCVs or TBVs.

During plant startup, once boiling commences, the RPV continues to heat up and pressurise to the rated pressure. The TBVs are initially used to control pressure with the RCS kept at saturated conditions. As power increases, the TCVs are gradually opened causing the TBVs to gradually close.

The RPV pressure is normally held constant by positioning the TCVs, so the turbine steam flow always follows the reactor. The TCVs are modulated by their control system to regulate the demand steam flow.

Further discussion on pressure control functions is provided in PSR Ch. 7, Section 7.3.

### 5.7.2 Safety Design Bases

Overpressure protection for the RCPB, as described in PSR Ch. 6, is provided through a combination of the NBS, the ICS, and SC1 I&C control system to limit peak pressure during AOOs to less than or equal to 110 percent of RPV design pressure as defined by the ASME BPVC, "Section III, Division 1, Article NB-7000, Overpressure Protection," subparagraphs NB-7120(b) and NB-7120(c) (Reference 5-38).

Refer to PSR Ch. 6, Section 6.2.1 for overpressure engineered safeguard functions.

### 5.7.3 Description

Reactor pressure is controlled from a cold vessel through pressurisation and heat up to power operation by the SC3 I&C control system for reactor pressure control using the TBVs and TCVs. The SC3 I&C control system during normal power operation controls reactor pressure using the TCVs. Refer to PSR Ch. 10, Sections 10.3 and 10.4.3 for normal operation and startup, respectively.

The SC3 I&C system IC Throttling Pressure Controller performs the functions to prevent exceeding the reactor cooldown rate. Detailed description is provided in PSR Ch. 7, Section 7.3.

### 5.7.4 Interfaces with Other Equipment or Systems

Figure 5-1 RCS Input-Output provides the RCS interfaces with other systems and equipment.

### 5.7.5 System and Equipment Operation

PSR Ch. 7, Section 7.3.3 describes normal pressure control provided by the TCVs. PSR Ch. 6, Section 6.2.1 under system design bases describes the overpressure engineered safeguard features and functions.

### 5.7.6 Instrumentation and Control

The SC1, SC2 and SC3 I&C Systems (refer to PSR Ch. 7, Section 7.3) provide control of RPV pressure using RCS pressure sensing instrumentation as described in Section 5.3.1 (RCS instrumentation subsystem).

**Leak Detection**

Means are provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. The components of the RCPB, including the ICS and RIVs are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with recognised codes and standards, and under an approved quality assurance program with approved control of records.

The BWRX-300 is a natural circulation design and has no pumps within the RCPB. Eliminating pump seal leakage significantly reduces anticipated leakage which could be identified. Additionally, the BWRX-300 does not make use of traditional safety or relief valves within the RCPB. The design simplification achieved with the BWRX-300 has eliminated many historical sources of identified reactor coolant leakage such that total leakage is reduced.

The leak detection methods employed inside, and outside containment. Leakage methods used for monitoring small unidentified leakage within containment are:

- Containment sump pump use
- Containment sump level and rate-of-rise
- Containment cooler condensate flow rate and increase
- Increase in radioactive fission product (noble gas and particulate) activity

These parameters are continuously monitored and/or recorded in the Main Control Room (MCR). The areas outside containment monitored for leakage are:

- The equipment areas in the RB
- The MS Tunnel area
- The TB
- The RWB

Leaks from line breaks are detected by monitoring the differential mass flow rate in CUW and SDC. The differential flow rate for CUW is measured in the supply lines from the vessel and discharge or overboard lines. The differential flow rate in the SDC is measured in the supply line that is connected to the ICS and the return or overboard lines, such that a SDC leak/break can be detected and isolated as needed.

The ambient temperatures in the MSL tunnel area, SDC area, CUW area, and FW heater area are monitored for indications of steam leakage. The steam tunnel is also monitored for area differential temperature. For room temperatures, the ambient temperature sensors are located near the ceiling, away from process piping and ventilation. Temperature elements used for differential temperature monitoring are located as near as possible to the area inlet and outlet ventilation flows.

Other leakage is detected by monitoring sump levels and flow rates from the plant sumps.

**Reactor Pressure Vessel Flange Leak Detection**

The RPV provides a flange leak detection connection to communicate to a groove between the two concentric O-ring seals such that leakage from the inner seal may be sensed as a buildup of pressure with the outer seal. The head closure is designed to ensure leakage is prevented from either seal with the design being capable of withstanding full reactor pressure out to the outer seal at all times. Seal effectiveness is determined by a closure head seal leak detection subsystem which monitors for leakage between the O-rings.

### 5.7.7 Monitoring, Inspection, Testing, and Maintenance

Section 5.11 describes the pre-service and ISI and system pressure test programs for USNRC Quality Group A, ASME BPVC, Division 1, Class 1 piping, and components. It describes these programs implementing the requirements of Subsection IWB of ASME BPVC-XI-1 (Reference 5-21). See Section 5.2.4 for discussion on degradation effects on materials.

### 5.7.8 Radiological Aspects

PSR Ch. 12, Section 12.3 (RCS) describes the various system and major component design considerations of the RCS related to plant radiation protection. PSR Ch. 12, Section 12.1.1 (Approach to ALARP) provides information on the design provisions and measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable in operational states and in accident or post-accident conditions.

### 5.7.9 Performance and Safety Evaluation

This section evaluates systems that protect the RCPB from over pressurisation. DL2 is an AOO state for which the plant is designed to limit pressure deviations and restore normal pressure. The evaluation of overpressure control in this section satisfies the AOO response. The bounding events for AOO pressure increase mitigation are described in PSR Ch. 15, Section 15.5.3.

The BWRX-300 Deterministic Safety Analysis, presented in PSR Ch. 15.5, is the analysis of record for responses to off-normal events and provides the information required to demonstrate that regulatory acceptance criteria are met. The selected bounding events resulting in the most significant challenges to the fission product barriers (including the RCPB) are summarised in Table 15.2-2. LOCA postulated initiating events where piping breaks occur in high-pressure pipes connected to the RPV, are discussed in Section 15.5.4 (Loss-of-coolant accidents - DBAs).

The use of an ICS design for BWRX-300 overpressure protection addresses the need for a reliable pressure mitigation which enhances the safety of the design. This is demonstrated by the safety analysis in PSR Ch. 15.5, Section 15.5.4 which concludes that there is no radiological release consequence for postulated pressure increase events.

The reactor integrity is assured by meeting the pressure criteria, provided in NEDC-34181P, “BWRX-300 UK GDA Ch. 15.3: Safety Objectives and Acceptance Criteria,” (Reference 5-39). The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction. The effect of this event does not result in any challenge to the temperature or pressure transient derived acceptance criteria for the fuel or pressure vessel. Therefore, these barriers maintain their integrity and function as designed.

## 5.8 Reactor Core Cooling Systems

### 5.8.1 Isolation Condenser System

The ICS is part of the RCPB and provides a DL barrier for overpressure due to AOO, DBA, and DEC events. If an MSL break occurs or the normal reactor cooling path is unavailable during an AOO or accident response, the ICS is credited for cooling and pressure protection of the RPV and the RCPB. The ICS is described in Section 5.1.9 and in more detail in PSR Ch. 6, Section 6.2.1.

### 5.8.2 Shutdown Cooling System

The BWRX-300 is a passive plant and does not have a traditional residual heat removal system. Removal of residual and decay heat is provided by the main condenser and the ICS (in conjunction with SDC). Shutdown Cooling (SDC) is available in Plant Modes 2 through 6. SDC is described in more detail in Section 9A.2.3.

## 5.9 Reactor Pressure Vessel Component Supports and Restraints

Support elements are provided for components included in the RCS and the connected systems.

### 5.9.1 Design Bases

**Safety Category 1 Functions**

- Ensure structural integrity of SC1 components when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events

**Safety Category 2 Functions**

- Ensure structural integrity of Safety Class 2 (SC2) components when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events

**Safety Category 3 Functions**

- Ensure structural integrity of Safety Class 3 (SC3) components when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events

**Safety Category N Functions**

- None.

### 5.9.2 Reactor Pressure Vessel Pedestal

The RPV pedestal is equipped with a RPV Bracket, where the RPV skirt is anchored using anchor bolts. The RPV pedestal provides structural support for the lower RPV Stabilizers and for the upper containment access platform module which supports the upper RPV Stabilizers. The RPV pedestal is a cylindrical, steel-plate composite structure that supports the RPV. The RPV pedestal, enclosed within the primary containment, is welded to a steel-plate composite basemat that is connected with the Steel-Plate Composite Containment Vessel. For additional description of the RPV pedestal refer to NEDC-34172P, “BWRX-300 UK GDA Ch. 9B: Civil Structures,” (Reference 5-40).

**RPV Pedestal Safety Basis**

The primary safety functions of the RPV pedestal are as follows:

- The RPV pedestal provides structural support to SSCs such as the RPV, RPV Stabilizers and miscellaneous platforms
- The RPV pedestal provides some radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOOs, DBAs, and DECs

The RPV pedestal is a SC1 and Seismic Category 1A structure.

The RPV pedestal does not serve any pressure-retaining function and is outside the scope of ASME Code applicability. Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and ISI of the BWRX-300 containment internal structures are listed in 006N3441 (Reference 5-32).

PSR Ch. 3 [Attachment 1], Section 3.6 discusses the structural integrity and/or functional integrity requirements of pressure-retaining components, their supports, and core support structures that are designed in accordance with the rules of the ASME Section III, Division 1.

### 5.9.3 Reactor Pressure Vessel External Supports

**Reactor Pressure Vessel Support Skirt**

The vessel support skirt is a truncated conical support appurtenance of the RPV. The support skirt is attached to the cylindrical shell of the vessel and located on the vessel shell exterior at an elevation above the center of gravity of the RPV assembly. The bottom of the skirt transmits the vertical vessel load to the vessel support pedestal. The pedestal is an internal PCS structure and is described in PSR Ch. 9B (Reference 5-40). The connection is designed to allow for radial movement due to thermal expansion and contraction of the vessel shell during operational heat-up and cool-down evolutions, respectively.

**Reactor Pressure Vessel Stabilizers**

The BWRX-300 has two sets of horizontal motion stabilizers with a set of six stabilizers located above the RPV bottom head and a second set of six stabilizers located below the MS nozzles. The stabilizer design is based on existing proven hardware design from BWR operating experience. The stabilizers are designed to the rules of ASME BPVC Section III, Division 1 (Reference 5-37) and are classified as SC1 and Seismic Category 1A. The main function of the stabilizers is to provide horizontal support to limit vessel motion by transferring load from the vessel to the adjacent civil structure. A total of twelve upper and lower vessel stabilizers function independently, depending on the direction of vessel motion due to an applied dynamic force, but horizontally and vertically parallel stabilizers act with combined resistance against vessel motion.

Refer to PSR Ch. 9B, Section 9B.2.2 for further details.

### 5.9.4 Reactor Pressure Vessel Internal Supports

**Chimney Support Brackets**

Lateral support brackets are located around the periphery of the chimney barrel exterior beneath the flange and attached at the vessel shell internal surface. These brackets brace the chimney and the chimney head – steam separator assembly against horizontal motion. The brackets are designed to the rules of ASME BPVC, Section III, Division 1 (Reference 5-37), classified as SC1, and Seismic Category 1A.

**Steam Dryer Support Brackets**

The steam dryer is supported by a set of four brackets equally spaced around the vessel shell internal wall. The dryer support ring rests on these brackets. Additional hold-down brackets are mounted inside the vessel head to restrain vertical movement of the dryer under seismic or dynamic loads. These hold-down brackets are installed so that they are over the lifting lugs of the steam dryer. Horizontal motion due to seismic or dynamic loads is limited by a set of seismic blocks mounted to the steam dryer support ring. The brackets are designed to the rules ASME BPVC, (Reference 5-37), classified as SC1, and Seismic Category 1A.

**Internal Piping Support Brackets**

*Reactor Water Cleanup System Internal Piping*

Two CUW internal suction lines are routed from flange connection inside the vessel to each respective nozzle, through inverted loops, down the vessel shell internal wall, and to the RPV bottom head. The piping for each internal suction line is attached to the vessel shell internal wall by brackets, spaced to provide support for anticipated loads and to inhibit pipe fatigue due to flow induced vibration. The brackets are designed to the rules of ASME BPVC, Section III, Division 1 (Reference 5-37), classified as SC1, and Seismic Category 1A.

*Head Vent Internal Piping*

The internal head vent piping inside the RPV closure head is a discharge line that is integrated into and through the bracket of one of the guide rods. The internal piping is split into a portion mounted inside the RPV closure head and a portion mounted below the vessel flange with an intersection of the internal piping at a guide rod bracket where the two ends are joined. The piping in the RPV closure head is mounted with brackets that hold the vent in place while allowing for thermal expansion and mitigating flow induced vibration effects. The brackets are designed to the rules of ASME BPVC, Section III, Division 1 (Reference 5-37), classified as SC1, and Seismic Category 1A.

### 5.9.5 Materials

Refer to Table 5-4 for reactor internal components materials listing.

### 5.9.6 Interfaces with Other Equipment or Systems

The BWRX-300 RPV pedestal and bioshield are integrated with the RB and PCS common mat foundation as shown in Figure 9B-1 and the RPV pedestal interfaces with the RPV through the RPV skirt support and horizontal stabilizers. Further description is provided in PSR Ch. 9B, Section 9B.2.2 (Interfaces with other equipment or systems).

### 5.9.7 Monitoring, Inspection, Testing, and Maintenance

RPV internal and external component supports are subject to an inspection program as described in Section 5.11. See Section 5.2.4 for discussion on degradation effects on materials.

### 5.9.8 Radiological Aspects

PSR Ch. 12, Section 12.3 (RCS) describes the various system and major component design considerations of the RCS related to plant radiation protection. PSR Ch. 12, Section 12.1.1 (Approach to ALARP) provides information on the design provisions and measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable in operational states and in accident or post-accident conditions.

### 5.9.9 Performance and Safety Evaluation

The RPV support design uses a conical skirt to support vertical load and horizontal stabilizers to minimise vessel motion due to horizontal load. The design is based on long operating experience with supports of the same basic design. The RPV support skirt design concept is a standard textbook method for mounting a pressure vessel that is employed in industries other than commercial nuclear power. The conical skirt shape permits the mid-vessel mounting of the skirt with the support ring foot set on the bracket attached to the cylindrical RPV pedestal. The support skirt is the primary interface with the civil structure supporting the RPV assembly and a transfer point for seismic accelerations from the RB to the RPV assembly. The RPV support skirt and stabilizer brackets are SA-508M, Grade 3, Class 1 material to match the RPV material. The support skirt is welded to the support skirt flange, which is integral to the RPV shell forging. The horizontal stabilizer assembly is predominantly fabricated from SA-105M carbon steel forgings.

The load of the BWRX-300 is more complex because of the relatively high locations of the vessel nozzles relative to the core and that there are large valves directly mounted to the vessel shell. Generally, BWR vessel load is concentrated below the vertical axis center point because of the mass of the core and the large liquid water volume in the lower vessel regions. However, the BWRX-300 has an additional large load concentration above the centerline because of the RIVs. The combined low and high mass centers result in a relatively higher RPV assembly mass centroid in the BWRX-300 design.

It is also known that RPV assembly response to dynamic or seismic accelerations in the horizontal direction is lowered by placing the attachment of the support skirt closer to the centroid of the RPV assembly mass “Pressure Vessel Design Handbook,” (Reference 5-41). The BWRX-300 RPV support skirt is attached at a mid-vessel elevation to place it much closer to the centroid than would be achieved with a traditional skirt attachment that is just above the bottom head. Because of the height of the natural circulation RPV (which is taller than a forced-circulation RPV due to the additional chimney component height) the BWRX-300 RPV is equipped with both upper and lower sets of horizontal stabilizers. The dual stabilizer sets design provides more control of vessel horizontal motion to limit vessel shell twisting or rocking around the skirt attachment. This limitation on motion also mitigates the dynamic input loads to the RPV nozzles to control the stress on the nozzle connections to the vessel shell. Further, limiting input motion to the RPV nozzles also limits the acceleration input into the RIV masses and provides greater assurance of reliable RIV function during postulated seismic and dynamic events.

The internal components support brackets are sized based on the static loads of the supported components. 'In-service monitoring, Tests, Maintenance and Inspections requirements' are defined in Section 13.3.2 of NEDC-34176P, “BWRX-300 UK GDA Ch. 13: Conduct of Operations,” (Reference 5-42).

## 5.10 Reactor Coolant System and Connected System Valves

The following information is consistent with 006N6121 (Reference 5-13). The BWRX-300 design focuses on the mitigation of LOCAs by reducing the number and size of RPV nozzles as compared to previous designs and limiting nozzle penetrations to the upper regions of the vessel. The RPV has integral isolation valves attached directly to the RPV as appurtenances for piping systems greater than DN20 nominal pipe diameter. The BWRX-300 design also includes an outboard MSCIV on each MSL.

### 5.10.1 Reactor Isolation Valves

The NBS RIVs, 004N9515 (Reference 5-12), and 006N6121 (Reference 5-13) are part of a fail-safe system designed to isolate the RPV during Service Levels A through D under the full range of reactor pressures and flows. The isolation system consists of the MSRIVs, FWRIVs, ICRRIVs, ICSRIVs, CUW RIVs and the RPV Head Vent RIVs. The RIVs are classified as SC1.

The RPV has integral isolation valves that are attached directly to and are appurtenances to the RPV. These valves limit the loss of coolant from large and medium size pipe breaks to mitigate the effect of LOCAs. The BWRX-300 design does not credit manual activation of the safety systems to mitigate the effects of a LOCA. The design of the RIVs consists of two valves in a single body that are independently able to isolate the respective line. RIVs are used on RPV connections to the MS subsystem, from the head vent to the supply and from the return of the ICS, to the CUW, and from the CFS.

The RIVs are designed in accordance with the rules and requirements of ASME BPVC-III NB, (Reference 5-6) and are Seismic Category 1B.

NBS RIVs are located inboard of each pipe greater than DN20 (3/4 in NPS) that is connected to the RPV. The RIVs are connected directly to the reactor vessel using bolted flange connections. Each NBS RIV assembly is connected to the outboard piping using bolted flange connections. The RPV nozzle automatic reactor isolation function is single failure proof in that there are two valves performing the same closure function.

Both RIVs on each respective nozzle perform the RPV isolation function to preserve reactor coolant inventory for large and medium pipe break LOCAs and also perform the inboard CIV function. Specifically, the inboard or outboard RIV for each respective nozzle performs the inboard CIV function. Both RIVs on each respective nozzle are designed and qualified to meet required testing criteria for the reactor isolation and containment isolation functions.

For the containment isolation function, either of the RIVs and the outboard CIV is required to close for each respective containment penetration.

The RIVs associated with ICS are fail-as-is type valves. The ICS RIVs are normally open during operation. Flow through the ICS during on-line operation is normally controlled by the ICS condensate return valves. Flow through the ICS is prevented by holding closed the parallel sets of ICS fail-open valves installed in the condensate return lines. Upon detection of a line break in an ICS train the actuator solenoids for the fail-as-is RIVs are energised by the SC1 I&C System to isolate the effected train. This function closes the RIVs on both the steam supply and condensate return lines for the affected train. The other two ICS trains are not isolated and remain available to perform their Safety Category 1 function. The RIVs can also be closed to isolate a train for maintenance and/or testing. This operational configuration is accounted for by the work planning in accordance with the plant Operational Limits and Conditions.

For information on SC1 I&C single failure proof design see 006N5114 (Reference 5-36). The RIVs with automatic closure requirements rely on SC1 battery-backed AC power. Electrical power utilised by the SC1 I&C System for automatic closure of the NBS ICS RIVs when warranted due to an identified ICS line break is derived from the battery backed electrical distribution system.

The RIVs are designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions. The RIVs are designed to withstand the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

The RIVs are equipped with OPEN and CLOSE position switches to provide position indication data to operators in the MCR. The RIVs are opened and closed via hydraulic actuators, most of which are spring-return type. An independent hydraulic system provides motive force for the RIV actuators.

The RIVs for ICS supply and return are designed to close in no greater than 5 seconds. The FWRIVs and FWCIVs are designed to close in no greater than 5 seconds. All BWRX-300 RIVs and MSCIVs have a proven low leakage potential. The RIV response time of each required division is to be verified within design limits per the IST program.

Figure 5-7 shows an example of RIV double valve bodies attached to the RPV using flange connections.

### 5.10.2 Main Steam Containment Isolation Valves

The BWRX-300 design includes an outboard MSCIV on each MSL. The MSRIVs and MSCIVs provide isolation of the MSLs for line breaks, MSCIVs for containment isolation, and when required during plant shutdown condition. The MSCIVs provide the DL3 Safety Category 1 isolation of the PCS in the event of accidents or other conditions and prevent the unfiltered release of containment contents that exceed appropriate limits. This containment isolation is single failure proof. The MSCIVs are fast-closing and fail-closed type valves. The MSCIVs are fast-closing and fail-closed type valves. These isolation valves outside of containment are located as close to the containment as practicable.

The MSCIVs are opened and closed via spring-return hydraulic actuators. An independent hydraulic system provides motive force for the MSCIV actuators.

The MSCIVs are designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions. The MSCIVs are designed to withstand the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

The MSCIVs are designed in accordance with the rules and requirements of ASME BPVC-III NCD (Reference 5-7) in accordance with their quality group classification. The MSCIVs are designed for Seismic Category 1B and certified by a qualification program prepared and performed in accordance with ASME QME-1 (Reference 5-31).

The MSRIVs and MSCIVs provide isolation of the MSLs. MSCIVs provide containment isolation, when required during plant shutdown conditions.

All CIVs have individual leakage testing performed to support the overall requirement to verify containment leakage is within the allowable limits. PSR Ch. 6, Section 6.5.6.1 describes containment isolation in detail.

### 5.10.3 Safety Design Bases

The RPV isolation function is designed as Safety Category 1 in accordance with the plant design requirement. All of the RIVs except those for the ICS nozzles have fail-closed type actuators with automatic isolation signals. The ICS nozzles must remain open during events requiring ICS to function and use fail-as-is actuators but have individual train automatic isolation on detection of a pipe leak or break in the train. The RIVs of each RPV nozzle are installed using flanged connections and mounted in a double valve body or two valves in a tandem configuration.

The RIVs also constitute the inboard CIV, where applicable, to provide the double isolation of the piping PCS penetration.

The ICS RIVs are normally open during operation. Isolation during on-line operation is normally controlled by the ICS condensate return valves. If called upon by an isolation signal, all other RIVs close. The ICS RIVs are designed to fail as-is, which is the normally open position. Flow will be initiated by the opening of the ICS condensate return valves. The only time the fail as-is position would be non-conservative is if the ICS RIVs are closed for maintenance but that would already be accounted for by the planning of the maintenance in accordance with the plant Operational Limits and Conditions.

The BWRX-300 design provides direct position indication of the RIVs. The ICS response time of each required division will be verified within design limits per the IST program. Figure 5-7 shows an example of RIV double valve bodies attached to the RPV using flange connections.

Design of the CIVs, with the exception of the RIVs, meets the rules and requirements of ASME BPVC, Section III, Division 1, Subsection NCD, Class 2 and 3 Components.

### 5.10.4 Materials

Materials are provided in Section 5.2.1. Material specifications used for the RIVs and the CIVs are in accordance with approved processes and with the rules and requirements of ASME BPVC, Section II (Reference 5-18).

### 5.10.5 Interfaces with Other Equipment or Systems

Figure 5-2 provides the interfaces with the RPV.

### 5.10.6 Instrumentation and Control

Each RCS MSRIV and MSCIV have automatic and manual control capability, with separate pilot valves, and independent test logic control.

The Safety Category 1 close function of the MSRIVs and MSCIVs is independent from the other control architecture.

Each RCS MSRIV and MSCIV is capable of isolating the MS piping in the event of a LOCA (or other events requiring containment or system isolation) to limit the release of reactor coolant.

### 5.10.7 Monitoring, Inspection, Testing, and Maintenance

RCS power-operated valves inside the RB are equipped for remote manual functional testing from the MCR. The ASME Class 1 valves undergo functional testing using equipment qualification methods that examine functionality and structural capacity under seismic and AOO loads, and accident loading conditions. Functional testing evaluates any identified leakage found during post-test inspections.

RCS is designed with provisions for initial and periodic testing of the ASME Code Class 1 and Class 2 system equipment, including hydrostatic test in accordance with the requirements of ASME BPVC, Section III, Division 1 and ASME BPVC, Section XI, Division 1. See Section 5.11 for further discussion on ISI and maintenance. See Section 5.2.4 for discussion on degradation effects on materials.

### 5.10.8 Radiological Aspects

PSR Ch. 12, Section 12.3 (RCS) describes the various system and major component design considerations of the RCS related to plant radiation protection. PSR Ch. 12, Section 12.1.1 (Approach to ALARP) provides information on the design provisions and measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable in operational states and in accident or post-accident conditions.

### 5.10.9 Performance and Safety Evaluation

All RCS RIVs fail-closed except for the ICS RIVs which fail as-is. The ICS RIVs fail as-is because they perform safety functions during DBAs. The ICS RIVs automatic closure function relies on the SC1 I&C system when initiated by the leak detection instrumentation. The RCS CIVs fail closed design provides conservative isolation of the PCS.

For small pipe breaks, the BWRX-300 design provides a substantial reservoir of water above the core that is sufficient to ensure RPV water level is maintained at or above a safe level following a LOCA. For larger pipe breaks, the respective RIVs, dependent on the location of the pipe break, will close rapidly to prevent significant loss of RPV inventory. In conjunction with RPV isolation, ICS will be actuated to maintain adequate core cooling. If the large break is located on an ICS line, the RIV associated with the train containing the line break will close. If the large break occurs inside containment, the MS CIVs and MSL RIVs will close to prevent unacceptable releases of containment contents.

The RPV RIVs and MSCIVs are single failure proof. The actuation signals for the RIVs are diverse. The RIV design includes redundancy in I&C components and features resulting in a worst-case single failure impacting only one RIV at each RPV nozzle. Diversity for penetrations where RIVs are credited as one of the CIVs have separate and diverse control systems that are single failure proof.

PSR Ch. 15, Section 15.5.4 evaluates the bounding DBAs. The scenarios for LOCA events are described in PSR Ch. 15, Section 15.5.4 (Loss-of-coolant accidents - DBAs) and conclude that during large break LOCAs, RPV inventory loss does not threaten fuel integrity. After RPV isolation, decay heat is removed by the ICS from the RPV. The LOCA analyses demonstrate that either the core remains covered, or fuel cladding temperature remains below the normal operating temperature for at least 72 hours using conservative assumptions for unisolated small break LOCAs. Therefore, fuel cladding temperature remains well below the fuel acceptance criteria, oxidation does not occur, and there is no hydrogen generation from cladding oxidation. The evaluation results for non-LOCA events demonstrate significant margin to the acceptance criteria, and barriers maintain integrity and functionality.

## 5.11 Access and Equipment Requirements for Inservice Inspection and Maintenance

The NBS is designed to provide adequate equipment removal paths and personnel access for maintenance, inspection, and replacement of components. The overall layout of mechanical equipment, piping, valves, and instrumentation is arranged so as not to interfere with operation and maintenance activities.

Pre-Service Inspection and ISI requirements for components and systems for the BWRX-300 are consistent with ASME BPVC-XI-1 (Reference 5-21). IST requirements for components and systems for the BWRX-300 are consistent with ASME OM (Reference 5-34). These Pre-Service Inspection, ISI, and IST requirements include examinations, inspections, and testing of the RPV and reactor coolant transporting systems, components, piping, and pipe supports, which are designed and installed in accordance with ASME BPVC, Section III, Division 1, and applicable codes in 006N3441 (Reference 5-32).

### 5.11.1 Accessibility

ASME BPVC, Section III, Class 1, 2, and 3 mechanical components and equipment (e.g., heat exchangers, pipe supports, pumps, valves, and vessels) are designed with accessible openings for ISI testing, which supports evaluations that justify the operational readiness of components and equipment.

Manufacturers of components and equipment, which require inspections and examinations to satisfy ASME BPVC, Section XI, Division 1 requirements, are examined by appropriate inspection and testing methods, as applicable per the ASME BPVC, Section III and ASME OM Code. The RPV engineering design includes provisions for the following ISI access requirements:

- Access to the annulus between the reactor pedestal wall and vessel is provided from below the vessel by provision in the bottom head insulation to access this region.
- Access to the RPV exterior above the RPV pedestal wall is provided by provision of removable insulation to access inspection regions.
- Removable insulation is provided on RCS piping and valves, which extends to the MSCIV.

### 5.11.2 Reactor Coolant Pressure Boundary Components

The components, which form the RCPB, are designed to permit the following:

- Periodic inspection and testing of important areas (such as pressure boundary welds and flanges during the post-outage leakage test) and features, to assess their leak-tightness and structural integrity.
- Establish a material surveillance program for the RPV.

### 5.11.3 Reactor Coolant System and Connected Systems

The RCS is defined as the components necessary to provide and maintain adequate core cooling conditions (pressure, temperature, and coolant flow rate) for the fuel in power operation. It includes the RPV, the MSLs, and FW lines up to and including the outermost CIV. These components form a major portion of the RCPB as described in Section 5.10.

Isolation and maintenance devices and valves are provided in the system to allow for maintenance of the RCS components. The following list contains the more important maintenance operations that must be performed on the RCS components:

- Servicing of RIVs and CIVs.
- Servicing and replacement of level, temperature, and pressure sensors.

- Servicing and replacement of solenoid elements in valve and pilot operators.
- Adjustment and servicing of valve operators, including limit and position switches.
- Cleaning drain line strainers.
- Adjustment and replacement of valve stem packing.
- Major overhaul of all power-operated valves and hydraulically operated valves.

Sufficient space, and the additional process system component facilities, is provided between penetration isolation valves and the PCS boundary to permit:

- Inservice inspection of non-isolable welds.
- Access and facilities to perform local leak rate testing of isolation valves.
- Access to operate local manual controls.
- Access to perform isolation valve assembly maintenance.
- Cutout and replacement of isolation valves using standard pipe cutting equipment, pipe fitting tools and equipment, and piping component welding equipment.

ASME BPVC, Section XI, Division 1, Class 1, and Class 2 RCS equipment is designed and arranged to provide ISI accessibility.

Areas requiring ISI inspection are provided with access spaces/routes and with removable and reusable insulation coverings.

### 5.11.4 Examination Categories and Methods

The design of each component and system considers the Non-Destructive Examination (NDE) method that is used to fulfill Pre-Service Inspection and ISI examination requirements. PSR Ch. 3, Section 3.10, describes requirements for in-service monitoring, testing, inspection, and maintenance programs for the BWRX-300. PSR Ch. 3, Section 3.10, states the requirements applied to components as designated in ASME BPVC (Reference 5-37). In addition, the design considers the operational and radiological concerns associated with the selected examination method to ensure that the performance of the required examination is practical during normal plant operation. The 3D layout of the plant includes acceptance criteria regarding access for inspection equipment and personnel. The examinations detect flaws, surface and volumetric discontinuities/imperfections, and original component position discrepancies, including conditions such as:

- Cracks, wear, laps, seams, cold shuts, laminations, physical damage.
- Corrosion type or erosion, debris.
- Leakages.
- Clearances, settings, physical displacements.
- Loss of integrity at bolted or welded connections.
- Missing or lost parts.

NDE methods provide the best technique for inspecting components and equipment that have been disassembled or installed during maintenance.

**Reactor Pressure Vessel Welds**

Reactor pedestal wall and the RPV insulation mounted around the vessel inside the pedestal wall provide access for remotely operated Ultrasonic Testing (UT) examination devices.

**Reactor Pressure Vessel Head, Studs, Nuts, and Washers**

The RPV head, studs, nuts, and washers are dry stored on the refueling floor during refueling operations. RPV studs, nuts, and washers are cleaned, lubricated, inspected, and reused, if acceptable.

Replacement or repair of RPV studs, nuts, and washers is subject to additional Repair/Replacement Plan requirements per the ASME BPVC, Section XI, Division 1, Article IWA4000. The design provides for removable insulation to ensure access for manual UT examinations of the RPV head welds.

RPV studs are volumetrically examined in place or when removed. RPV nuts and washers are accessible for visual examinations.

**Bottom Head Welds**

Access to the bottom head-to-shell weld is provided from the under-vessel area through removable insulation around the bottom head, the CRD housings and nuclear instrument housings of the RPV. This design provides access for manual or automated UT examination equipment.

These welds are required to be accessible in order to perform ASME BPVC, Section XI, Division 1, VT-2 examinations during system leakage testing in accordance with ASME BPVC, Section XI, Division 1, Table IWB-2500-1.

Access to welds inside the bottom head is made through the respective housings during under-vessel maintenance work. Access is provided for nozzle weld examinations.

**Reactor Vessel Support**

Access is provided for visual examination of the RPV support structure.

**Piping, Valves and Supports**

Design and physical arrangement of piping, valves, and supports provide personnel access to each weld location to perform NDE.

Working platforms facilitate servicing of valves and supports. Platforms and ladders provide access to piping welds, including the pipe-to-reactor vessel nozzle welds.

Removable thermal insulation is provided on welds and components, which require access for examination or are located within high radiation areas.

Welds are located to permit UT examination from at least one side. A 100 percent volumetric in-service examination of all pipe welds is conducted during each inspection interval as defined within ASME BPVC, Section XI, Division 1, Sub-article IWA-2400.

**Augmented In-Service Examinations**

*High-Energy Piping*

RCS piping located between the CIVs, and extending to the Seismic Interface Restraint, is seamless (no longitudinal welds). After installation, circumferential welds in the aforementioned piping are subjected to 100 percent Radiographic Testing in accordance with ASME BPVC-III NCD (Reference 5-7).

*Flow Accelerated Corrosion Examinations*

ASME BPVC, Section III, Division 1, Class 1, 2, and 3 piping systems, including piping and components, may be susceptible to FAC. This wear mechanism compromises the structural integrity (wall thinning) of high-energy carbon steel piping systems and components.

**Non-destructive Examination Methods**

NDE methods (surface, visual, and volumetric examinations) are described within ASME BPVC, “Section V, Non-destructive Examination,” (Reference 5-43), ASME BPVC-XI-1 (Reference 5-21), and applicable codes and standards in 006N3441 (Reference 5-32).

For ASME BPVC, Section III, Division 1, Class 1 and Class 2 welds, UT or Radiographic Testing examinations are selected for ISI examinations.

Radiographic Testing examinations are primarily used as a volumetric method. In addition, Radiographic Testing examinations supplement UT examinations to improve coverage of the required examination volume.

*Surface Examination Methods*

Surface examinations of welds and bolts indicate the presence of surface discontinuities, which are performed by using Eddy Current Testing, Liquid Penetrant Testing, Magnetic Particle Testing, or UT methods.

Any linear flaw detected within a component by non-destructive techniques, which exceeds the allowable linear surface flaws standards is recorded.

*Visual Examination Methods*

Visual examinations indicate the presence of surface discontinuities and imperfections, leakage from pressure-retaining components, and the general mechanical and structural condition of components.

*Volumetric Examination Methods*

Volumetric examinations indicate the presence of discontinuities throughout the volume of the material, which are performed by using an Acoustic Emission, Eddy Current Testing, Radiographic Testing, or UT method. Volumetric examinations are used for the examination of welds of vessels, piping, studs, and bolts.

### 5.11.5 Inspection Intervals

The ISI examination and system pressure test intervals, and percentage of examinations should conform with ASME BPVC, Section XI, Division 1, Sub-articles IWB-2400, IWC-2400, IWD-2400, IWF-2400, ASME OM Code, Subsections ISTC, ISTD, ISTF, and applicable codes and standards in 006N3441 (Reference 5-32). Inspection intervals are a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages.

MSRIVs and MSCIVs are periodically local leak rate tested to meet the leak rate acceptance criteria in accordance with the design requirements.

The MSRIVs and MSCIVs are diagnostically tested to verify that the MSRIVs and MSCIVs are properly adjusted to perform their respective safety MSL isolation functions, and to monitor for performance degradations.

MSL isolation trip channels are tested periodically to verify that MSRIV and MSCIV control pilots respond, and position instrumentation is capable of providing required position signals to the respective control systems for required alarms and automatic trip functions.

MSRIV and MSCIV leakage rate testing is based on MSL cumulative allowable leakage rate established by DBA dose evaluation. Individual valve leak rate testing for diagnostic and rework/refurbishment is based on the design performance criteria.

FWRIVs leakage rate testing is based on total allowable leakage established by design analysis for PCS total leakage. Individual valve leak rate testing for diagnostic and rework/refurbishment is based on the design performance criteria.

RCS power-operated valves inside the RB are capable of remote manual functional testing. These valves are designed to be tested from the MCR. This testing verifies that proper position indication is displayed in the MCR.

### 5.11.6 Provisions for Evaluating Examination Results

Examination results are evaluated and accepted in accordance with ASME BPVC, Section XI, Division 1, Articles IWA-3000, IWB-3000, IWC-3000, IWD-3000, and IWF-3000, with repairs based on the requirements of IWA-4000, and applicable codes and standards listed in 006N3441 (Reference 5-32).

Recorded results meet the acceptance standards specified within ASME BPVC, Section XI, Division 1, Sub-articles IWB-3400 and IWB-3500, IWC-3400 and IWC-3500, IWD-3400 and IWD3500, IWF- 3400, and applicable codes and standards listed in 006N3441 (Reference 5-32).

Components containing flaws or relevant conditions and accepted for continued service are subjected to successive period examinations.

### 5.11.7 System Pressure Tests

RCS is designed with provisions for initial and periodic testing of the ASME BPVC, Section III, Division 1, Class 1 and Class 2 equipment, including hydrostatic testing.

ASME system pressure tests (hydrostatic test, leakage test or pneumatic tests) are performed in accordance with ASME BPVC, Section XI, Division 1, Articles IWA-5000, IWB-5000, IWC-5000, IWD-5000, and applicable codes and standards listed in 006N3441 (Reference 5-32) to ensure leak tightness of components, equipment, mechanical and welded joints, integral support-piping attachment welds (i.e., lugs), and piping.

ASME system pressure-retaining boundaries are defined within ASME BPVC, Section XI, Division 1, Paragraphs IWA-5220, IWB-5222, IWC-5222, and IWD-5222.

VT-2 examinations are performed in accordance with ASME BPVC, Section XI, Division 1, Paragraphs IWA-2212, and applicable codes and standards listed in PSR Ch. 1, Appendix B.

System hydrostatic test for RCPB systems is performed at a pressure and temperature, which corresponds to ASME BPVC, Section XI, Division 1, Sub-article IWB-5230, Table IWB-5230-1, ASME OM Code, Subsections ISTC, ISTF, and applicable codes and standards listed in 006N3441 (Reference 5-32).

### 5.11.8 Surveillance Program Requirements and Overview

The surveillance programs that are established address the several aspects of the materials used in the BWRX-300. PSR Ch. 3, Section 3.10 provides additional information. Specifically, there is a need for:

- Monitoring the integrity of the pressure-retaining RPV and its attachments.
- Monitoring the time limiting aging impact of the RPV.
- Monitoring of the integrity of the reactor internals.
- Monitoring the piping components for degradation.
- Monitoring the water chemistry program for its mitigation effectiveness.
- Monitoring for FAC.

A discussion of the key elements of each surveillance program is presented in the next sections.

**Pressure-Retaining Inservice Inspection Program**

There are ISI requirements based on the ASME BPVC, Section XI, Division 1 for pressure retaining components and their integral attachments in the BWRX-300. This program serves as a model for the surveillance program that will include periodic visual, surface, and/or volumetric examination and leakage tests of the pressure-retaining components including the RPV, nozzles and attachments. The surveillance program will include specific scope for each of the pressure retaining components. The program also will include defined preventative actions which include adequate methods to detect degradation that will occur with plant aging. The program will include defined monitoring and trending aspects, and specific correction action plans to address degradation identified. Finally, the program will track operating experience in these pressure retaining components and use this information to update the ISI program.

**RPV Material Surveillance Programs**

The material surveillance program monitors $K_{IC}$ property changes of ferritic materials in the RPV beltline region resulting from exposure to neutron irradiation, including thermal information requirements pertaining to materials and surveillance capsules. The RPV material surveillance specimens are provided in accordance with requirements of the ASTM E185 standard and contain materials representative of the materials and material conditions used in the RPV beltline region. The plan accounts for the entire lifetime of the plant.

**Boiling Water Reactor Vessel and Internals Project (BWRVIP)-Type Reactor Internals Surveillance Inspections**

The reactor internal components are susceptible to aging degradation mechanism, particularly SCC and for the core shroud, and IASCC. While efforts are made to minimise degradation through material selection, component fabrication and the use of Hydrogen Water Chemistry/OLNC, the BWRVIP has developed inspection programs to monitor components for the impact of environmental degradation during plant operation in current operating BWRs in key internal components. NUREG-1801 (Reference 5-25) provides an overview of these programs. These existing programs are used as the initial basis to develop BWRX-300 surveillance inspection plans for the different reactor internal components. The BWRVIP program includes inspection and flaw evaluation approaches that provide assurance of the long-term integrity. The guidelines provide information on component description and function, susceptible locations, and safety consequences of failure. They also provide recommendations for methods, extent, and frequency of inspection. The programs are focused on managing the effects of cracking due to SCC, IGSCC, or IASCC, cracking due to fatigue, and loss of toughness due to neutron and thermal embrittlement. The BWRVIP program includes components such as the core shroud, core plate, core shroud support, top guide, core plate, steam dryer, and CRD housings.

**BWR Piping and Safe End Welds/HAZs Surveillance Inspections**

The BWRVIP following USNRC guidance has developed inspection programs to monitor piping weld regions for the impact of environmental degradation, particularly IGSCC, during plant operation. These programs augment the ASME BPVC, Section XI, Division 1, ISI program to address IGSCC concerns. Section 5.2.6 discusses the key measures that are taken to significantly reduce susceptibility of BWRX-300 piping components to SCC. The existing BWRVIP programs for ISI can be used as the initial basis to develop the BWRX-300 surveillance inspection plans for the different stainless steel piping welds/HAZ regions to minimise any SCC risks for degradation over the plant operating lifetime.

**Water Chemistry Monitoring/OLNC Effectiveness**

The NUREG-1801 (Reference 5-25) establishes that water chemistry is key to the BWR aging management plan for operating BWRs. This will also be the case for the BWRX-300. The

BWRX-300 will comply with the BWR Water Chemistry Guidelines which address parameters for sample, frequency, and action levels. The guidelines provide proactive water chemistry guidance for mitigating environmentally assisted corrosion, maintaining fuel integrity, controlling FAC and controlling radiation fields.

**Flow Assisted Corrosion Monitoring**

For carbon steel piping, the FAC surveillance program defines key piping inspection locations, to evaluate and monitor wall thinning.

### 5.11.9 Program and Milestone Implementation

Refer to PSR Ch. 13: Conduct of Operations, Section 13.3.3.

## 5.12 Reactor Auxiliary Systems

### 5.12.1 Chemical and Inventory Control Systems for the Reactor Coolant

IGSCC can occur in BWR startup and at-power environments. The normal BWR environment during power operation is 286°C water containing dissolved oxygen, hydrogen, and small concentrations of ionic and non-ionic impurities. Increasing levels of many ionic impurities influence SSC behaviour of RCPB materials and may also affect fuel performance. Reactor water conductivity is maintained at or below the defined limits.

Besides being a major contributor to IGSCC of sensitised stainless steels, reduction of oxygen content is known to reduce the tendency for pitting and cracks of most plant materials. During power operation, most of the oxygen content of reactor water is due to the radiolysis of water in the core and, therefore, oxygen control cannot be achieved through traditional chemistry and operational practices. Reactor water oxygen control to low, plant-specific levels can be obtained through hydrogen injection from a Hydrogen Water Chemistry System. Hydrogen Water Chemistry is an established technique for mitigating and reducing the growth rates of IGSCC in reactor vessel internals. Control of reactor water oxygen during startup/hot standby is accomplished by utilising the deaeration capabilities of the condenser.

In order to reduce the risk of IGSCC in reactor vessel internals, the BWRX-300 plant chemistry regime includes Hydrogen Water Chemistry System and OLNC, see PSR Ch. 23, Sections 23.1.3.1 and 23.1.3.2, respectively. The OLNC operates in conjunction with the Hydrogen Water Chemistry System; although, they are both completely separate systems which have no mechanical or electronic interrelationships.

The OLNC provides a means for the injection of a noble metal salt solution directly into the reactor coolant flow path. In combination with Hydrogen Water Chemistry injection, the noble metal deposition in the reactor provides a catalyst effect on vessel surfaces to facilitate the recombination of free hydrogen and oxygen molecules to minimise the oxygen available to initiate or encourage IGSCC crack growth. Based on the catalytic surface recombination efficiency provided by OLNC, less hydrogen is required to mitigate IGSCC initiation and reduce IGSCC crack growth rates. Due to the requirement for lower hydrogen, MSL dose rates are also lower.

The BWRX-300 water chemistry sampling and monitoring program is designed to analyze and monitor system chemistry for trending with alarm notification so actions can be taken to stay within operating specifications. The water chemistry control parameters, recommended operating limits, and recommended monitoring frequencies are developed to minimise the potential for IGSCC by controlling both ionic impurity and oxidizing radiolysis product concentrations in the reactor water. The Chemistry Control Program supports minimizing corrosion from chemical contaminants and monitoring chemical additives used to limit corrosion and contamination buildup. PSR Ch. 13, Section 13.3.1.3 (Fitness for service) provides further details.

### 5.12.2 Reactor Water Cleanup System

The following information is consistent with 006N7609 “BWRX-300 Reactor Water Cleanup System (G31 SDD),” (Reference 5-44). The CUW provides blowdown-type cleanup flow for the RPV during the reactor power operating mode. Actual cleanup or filtration and ion removal is performed by the condensate system. Additionally, it provides an overboarding flow path to the condenser hotwell (condensate pump suction) or liquid radwaste directly from the RPV lower region. The CUW system is described in more detail in PSR Ch. 9A, Section 9A.2.2.

### 5.12.3 Residual Heat Removal System

The ICS removes decay heat after any reactor isolation and shutdown event during power operations when the main condenser is not available (see section 5.8 and PSR Ch. 6, Section 6.2.1).

The ICS is an ECCS that also provides RPV overpressure protection for which further details can be found in PSR Ch. 6, Section 6.2.1.

### 5.12.4 High Point Vents of the Reactor Coolant System

The RPV head vent subsystem includes piping internal to the RPV head, two flange-mounted in-series RIVs, and piping to the MSL or Quench Tank, both located inside PCS. This vents non-condensable gases from the reactor steam dome via a connection to one of the MSL during plant operation. It permits the gases to be released from the RPV so that the RPV can be filled with water for hydrostatic testing and provides the upper tap for RPV level measurement during reactor shutdown. In addition, the horizontal portions of the head vent line piping to either the Quench Tank or the MSL have a downward slope in the direction of normal flow ensuring that no condensate stays in the line and prevents hydrogen from collecting.

A vacuum breaker is provided to equalise head vent line pressure with the PCS following cessation of flow through the vent line to the Quench Tank. The vacuum breaker is installed to prevent bypass leakage from the vent line to the PCS, or deposition of crud or debris in the vacuum breaker from the outlet side. The RPV head vent vacuum breaker is located at a high point near the downstream side of the Quench Tank pipe branch outboard isolation valve.

During reactor shutdown, reactor water level can be measured using the vent connection as the upper tap for the level instrumentation sensor. Also, during shutdown the RCS head vent connection permits air and non-condensable gases to be released from the RPV into the Quench Tank inside the PCS. The RCS head vent allows air to enter the RPV through the head vent connection when draining the RPV. During at-power operation, the RCS head vent connection permits, and non-condensable gases to be released from the RPV into the MSL.

The non-condensable gases are extracted by differential pressure between the reactor head and the steam lines and then swept from the steam lines to the condenser.

### 5.12.5 Control Rod Drive System

The CRD system contains components which form part of the RCPB, along with components which are important to safety to shut down the reactor. Those portions of the CRD system are classified as DL3/SC1.

The RPV provides integral CRD housing flanges for mounting the FMCRD to the vessel. The housing provides guidance and support for the FMCRDs in support of the normal control rod movement and scram functions. The RPV also provides mechanical interface at the coupling between the FMCRD and control rod, and at the coupling between the FMCRD and control rod guide tubes. These mechanical interfaces prevent ejection of a control rod in the event of a postulated failure of a CRD housing.

The FMCRDs provide electric-motor driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods during off-normal conditions. The FMCRD major components that are part of the RCPB are analyzed and evaluated for the Service Level D conditions in accordance with ASME BPVC.

The CRD system materials specifications are discussed in PSR Ch. 4, Section 4.5. The design of the CRD system is further described in PSR Ch. 4, Section 4.5.

**Table 5-1: List of Interfacing Systems with RCS**

| MPL Code | System |
|---|---|
| B21 | Nuclear Boiler System |
| D11 | Process Radiation and Environmental Monitoring System |
| E52 | Isolation Condenser System |
| F15 | Refueling Equipment and Servicing |
| G12 | Control Rod Drive System |
| G31 | Reactor Water Cleanup System |
| J11 | Core and Fuel |
| N21 | Condensate and Feedwater System |
| N31 | Main Turbine Equipment |
| N35 | Moisture Separator Reheater System |
| N37 | Turbine Bypass System |
| N61 | Main Condenser and Auxiliaries |
| P52 | Plant Pneumatics System |
| R20 | Standby Power System |
| R30 | Preferred Power System |
| R41 | Grounding and Lightning Protection System |
| T10 | Primary Containment |
| U50 | Equipment and Floor Drain System |
| U71 | Reactor Building Structure |
| U72 | Turbine Building Structure |

**Table 5-2: Typical Materials for Reactor Pressure Vessel Components**

| Components | Form | Material Type | Typical ASME/ASTM Specification |
|---|---|---|---|
| RPV Shells and Heads | Plate<br><br>Forging | Mn-$^1/_2$Mo-$^1/_2$ Ni Low Alloy Steel<br>$^3/_4$ Ni-$^1/_2$Mo-Cr-V Low Alloy Steel | SA-533M Type B, Class 1<br><br>SA-508M Grade 3, Class 1 |
| RPV Nozzle Safe Ends | Forging | Low Alloy Steel | SA-508M Grade 3, Class 1 |
| RPV Main Closure Bolting, Flange Bolting | Bolting | Low Alloy Steel | SA-540M Grade B23 or B24, Class 3 |
| Low Alloy Steel Piping | Seamless Forged & Bored | Low Alloy Steel | SA-336M Grade P22, SA-369M FP22 |
| Carbon Steel Piping | Seamless | Carbon Steel | SA-333M Grade 6 |
| Carbon Steel Welds | Covered Electrode or Filler metal | Carbon Steel | SFA-5.1 SFA-5.18 |
| Low Alloy Steel Welds | Covered Electrode or Filler metal | Low Alloy Steel | SFA-5.5<br>SFA-5.23<br>SFA-5.28 |
| Low Alloy Steel Piping Welds (2-1/4 Cr 1 Mo) | Covered Electrode or Filler metal | Low Alloy Steel (2-$^1/_4$ Cr 1 Mo) | SFA-5.5 SFA-5.28 |
| Stainless Steel Welds | Covered Electrode or Filler metal | Stainless Steel | SFA-5.4<br>SFA-5.9 |
| Nickel Alloy Welds | Filler Wire | Nickel Alloy | SFA-5.14 |

**Table 5-3: Typical Compositions of Low Alloy Steel and Carbon Steel Materials**

| | C | Cr | Mn | Mo | Ni | Fe | P | S | Si | Other |
|---|---|---|---|---|---|---|---|---|---|---|
| SA-533M, Gr. B (Notes 1 & 5) | .25 max | | 1.15-1.5 | .45 - .60 | .40 - .70 | Rem | .025 max | .025 max | .15 - .40 | |
| SA-508M, Grade 3 (Note 1) | .25 max | .25 max | 1.20 - 1.50 | .45 - .60 | 0.40-1.00 | Rem | .025 max | .025 max | .40 max | Restricted V, Cu, Nb, Ca, Cu, B |
| SA-540M Grade B23 (Notes 2 & 3) | .37-.44 | .65-.95 | .6-.95 | .2-.3 | 1.55-2.00 | Rem | .025 max | .025 max | .15-.35 | |
| SA-333M Grade 6 Carbon Steel (Note 4) | 0.30 max | 0.3 max | 0.29-1.06 | 0.12 | 0.4 max | Bal. | 0.025 max | 0.025 max | 0.10 (min) | - |
| SA-335M Grade P22 (2 ¼ Cr-1 Mo) | 0.5-0.15 | 1.9-2.6 | 0.3-0.6 | 0.87-1.13 | - | Bal. | 0.025 max | 0.025 max | 0.5 max | - |

Notes:

(1) There are special controls of key elements and interstitials to address different embrittlement processes including radiation embrittlement and reduction of toughness in the RPV.

(2) Maximum allowable yield strength 1034 MPa (150 psi).

(3) Composition ranges shown do not include the Over/Under Production Variation values as stated in SA-504M

(4) There are also composition controls to address FAC that are applicable to standard carbon steel piping. These mechanisms are discussed in the next sections.

(5) The element ranges listed are applicable to the heat analysis per specification SA-533M.

**Table 5-4: Typical Reactor Internal Components and Materials**

| Component(s) | Form | Material Type | Typical ASME/ASTM Specification |
|---|---|---|---|
| Steam Dryer | Plate, Bar or Forgings | Stainless Steel | SA-240M, SA-182M, SA-479M<br>Type 304/304L, 316/316L |
| Core Shroud and Support Ring | Plate or Forgings | Stainless Steel | SA-240M, SA-182M<br>Type 316L |
| Core Support Structure | Plate or Forgings | Stainless Steel | SA-240M, SA-182M<br>Type 316L |
| Core Support Legs | Plate, Bar or Forgings | Columbium modified<br>Ni-Cr-Fe (A600M)<br>Nickel Base Alloy | SB-168, SB-166 or SB-564 Nickel Alloy modified per<br>ASME Code Case N-580-2 |
| Top Guide | Plate or Forgings | Stainless Steel | SA-240M, SA-182M<br>Type 304/304L, 316/316L |
| Chimney | Plate or Forgings | Stainless Steel | SA-240M, SA-182M<br>Type 304/304L, 316/316L |
| Control Rod Drive Housings | Seamless Pipe or Forgings | Stainless Steel | SA-312M, SA-182M<br>Type 304/304L, 316/316L |
| Fuel Supports | Castings | Stainless Steel | SA-351M<br>Grade CF3 |
| In-core Housings | Seamless Pipe or Forgings | Stainless Steel | SA-312M, SA-182M<br>Type 304/304L, 316/316L |
| Control Rod Drive<br>Penetration Stub Tubes | Bar, Pipe or Forging | Ni-Cr-Fe Alloy | SB-166, SB-167 or SB-564<br>Nickel Alloy 600 modified per<br>Code Case N-580-2 |
| Core Plate Bolts | Bar | XM-19 | SA-479M Type XM-19 |
| Stainless Steel Welds | Covered Electrode or Filler metal | Stainless Steel | SFA-5.4<br>SFA-5.9 |
| Nickel Alloy Welds | Filler Wire | Nickel Alloy | SFA-5.14 |

**Table 5-5: Typical Compositions: Standard BWR-300 Materials and Special Compositional Controls for Toughness and IASCC Resistance**

| | C | Cr | Mn | Mo | Ni | Fe | P | S | Si | Other |
|---|---|---|---|---|---|---|---|---|---|---|
| 316L | 0.030 | 16-18 | 2.00 | 2-3 | 10-14 | Bal. | 0.045 | 0.030 | 0.75 | - |
| 304L | 0.030 | 17.5-19.5 | 2.00 | - | 8-12 | Bal. | 0.045 | 0.030 | 0.75 | - |
| 316NG (316L plus N) | 0,020 | 16-18 | 1.0-1.5 | 2-3 | 12-14 | Bal. | 0.025 | 0.01 | .25 | N (0.06-0.10) |
| Alloy 600 | 0.15 | 14-17 | 1.0 | - | 72 (min) | 6-10 | - | 0.015 | 0.5 | Cu 0.5. |
| Alloy 600M* | 0.050 | 14-17 | 1.0 | - | 72 (min) | 6-10 | - | 0.015 | 0.5 | Cu 0.5<br>Nb 1-3 |
| Alloy X750 | 0.08 | 14-17 | 1.00 | 0.7-1.2 | 70 min. | 5-9 | - | 0.01 | 0.5 | Ti 2.25-2.75<br>Al 0.4-1.0 |
| Alloy 718** | 0.08 | 17-21 | 0.035 | 2.8-3.3 | 50-55 | Bal | 0.015 | 0.015 | 0.035 | Nb, Ti, Al, Co |
| XM-19 | 0.04 | 20.5-23.5 | 4-6 | 1.5-3 | 11.5-13.5 | Bal. | 0.045 | 0.030 | 1.00 | Nb 0.1-0.3; V 0.1-0.3 |
| Alloy 17-4PH | 0.07 | 15-17.5 | 1.00 | - | 3-5 | Bal. | 0.040 | 0.030 | 1.00 | Cu 3-5 |

Notes:

(1) * Nb-modified Alloy 600 per BPVC Case N-580-2.

(2) **Additional constituents: Nb: 4.75-5.5; Ti: 0.65-1.15; Al: 0.2-0.8; Co: 1.0 max.

(3) All single values are maximums, unless otherwise specified.

(4) Controlled Residuals for IASCC Resistance - Type 316L: Silicon controls: Silicon in the core shroud and top guide shall be limited to 0.10 percent maximum.

(5) Controlled Composition for Improved Weldability - Stainless Steel: Composition controls to keep ferrite levels >3%.

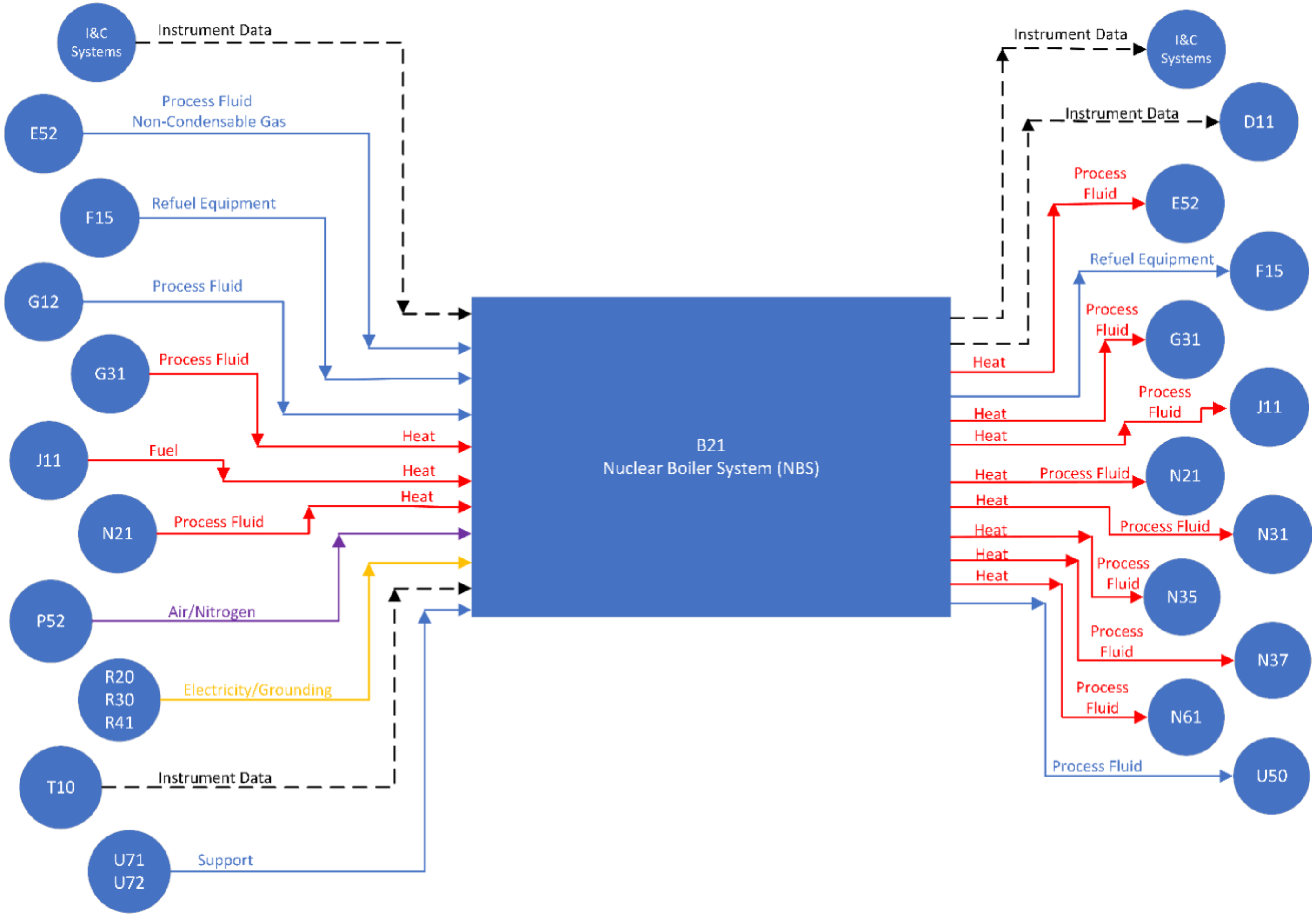


**Figure 5-1: RCS Input – Output Diagram. For system codes see Table 5-1.**

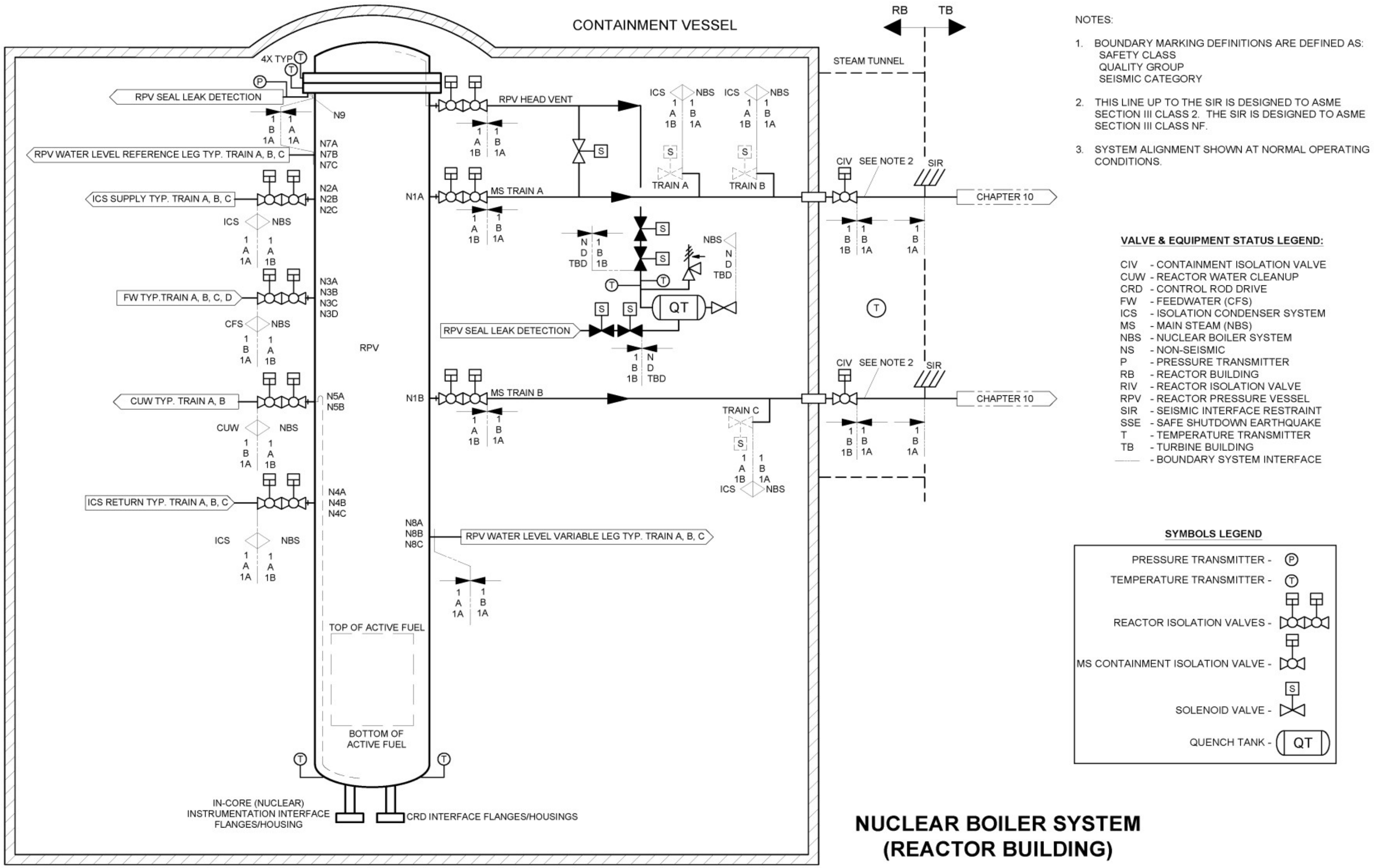


**Figure 5-2: Reactor Coolant System Simplified Diagram (Located in the Reactor Building)**

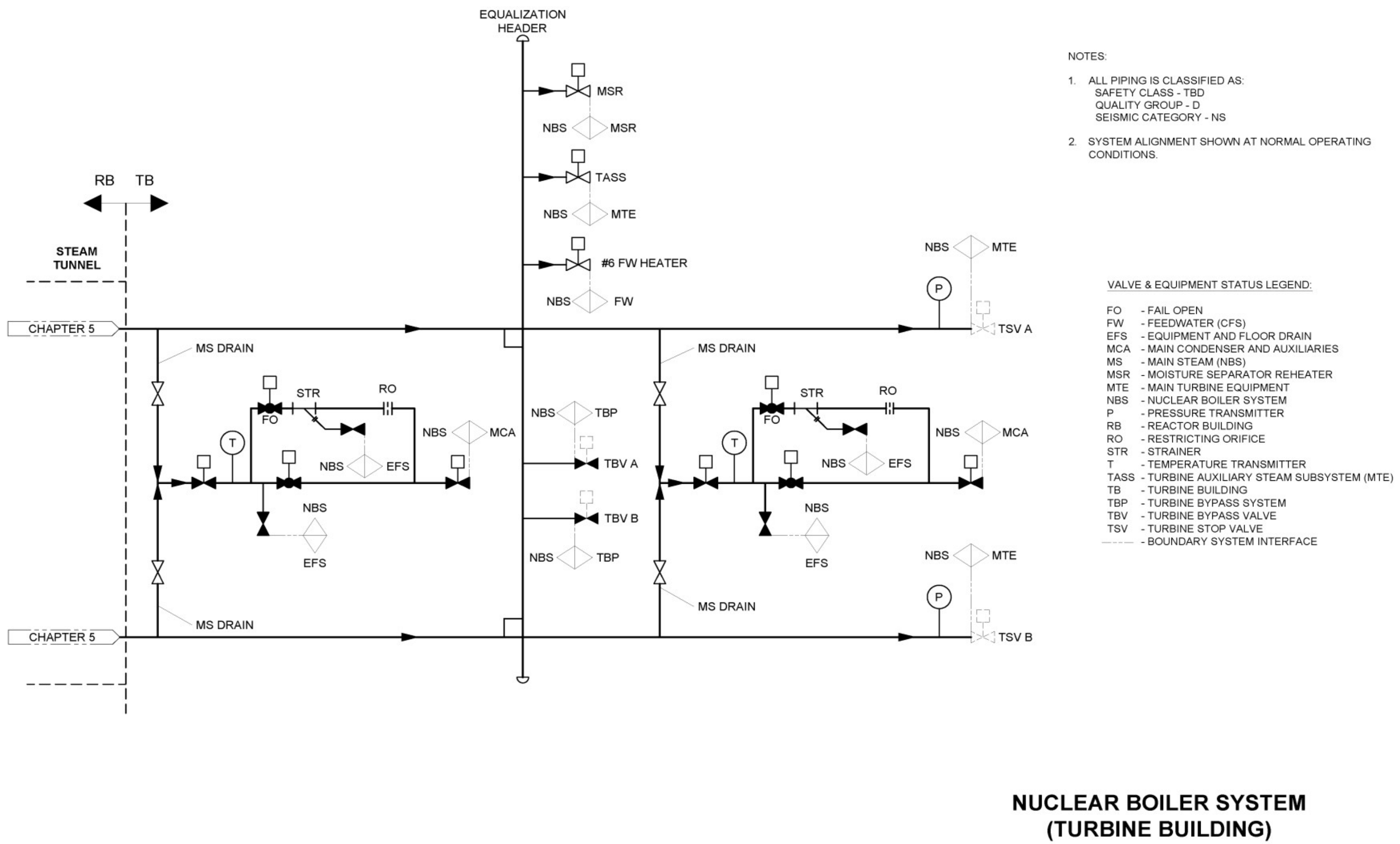


**Figure 5-3: Reactor Coolant System Simplified Diagram (Located in Turbine Building)**

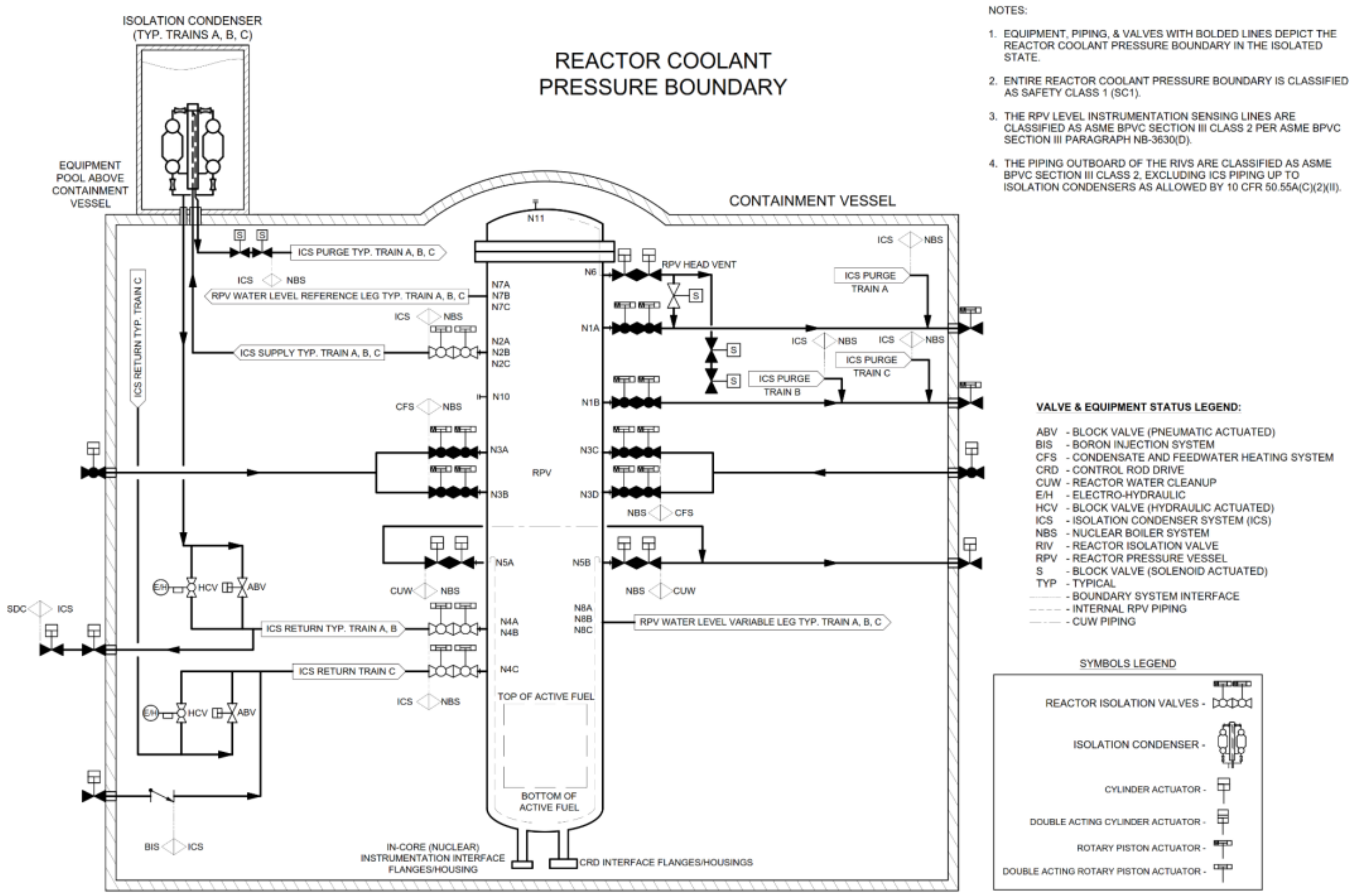


**Figure 5-4: Reactor Coolant Pressure Boundary**

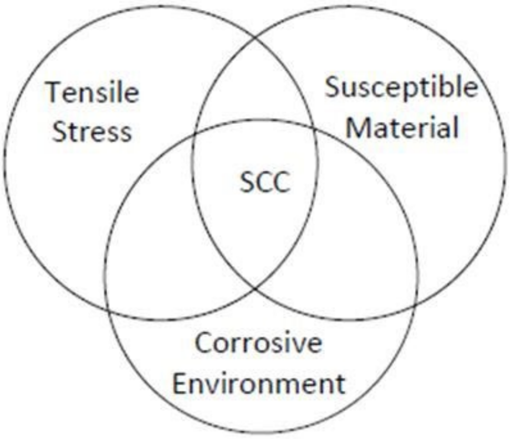


**Figure 5-5: The Three Necessary Factors to Produce Stress Corrosion Cracking**

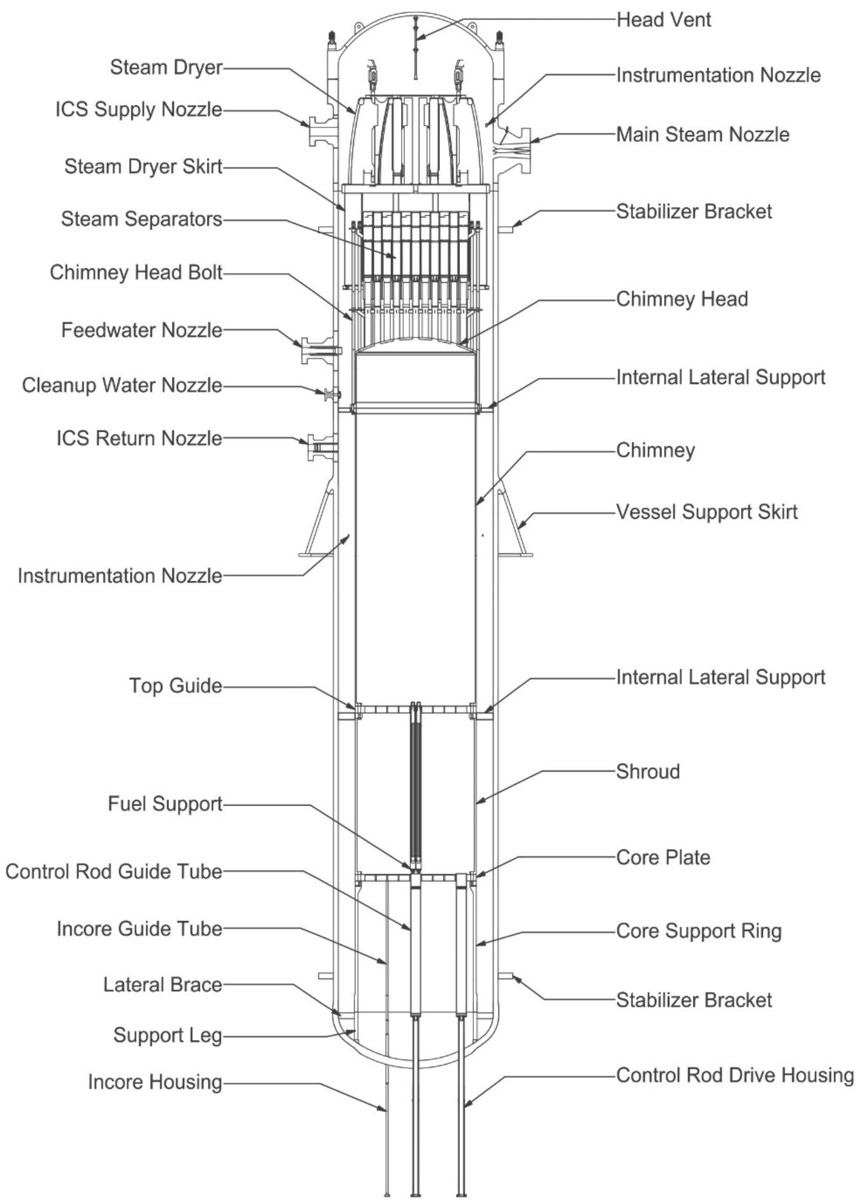


**Figure 5-6: BWRX-300 Reactor Pressure Vessel and Internals**

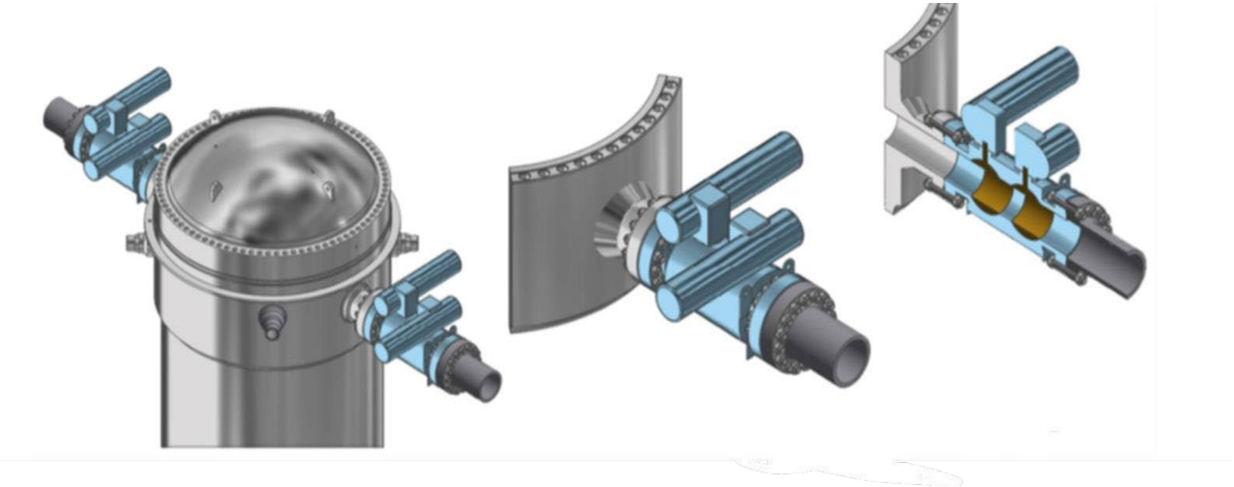

**Figure 5-7: Reactor Pressure Vessel Isolation Valve Assembly (Example)**

## 5.13 References

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# APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

The ONR "Safety Assessment Principles (SAPs)," 2014 (Reference 5-45) identify ONR's expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The Claims, Arguments and Evidence (CAE) approach can be explained as follows:

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 GDA Safety Case Development Strategy," (Reference 5-46) 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 Reference 5-46 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 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-300 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.

**Risk Reduction ALARP**

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:
  - RGP has demonstrably been followed
  - OPEX has been taken into account within the design process
  - All reasonably practicable options to reduce risk have been incorporated within the design
- It supports its applicable level 3 sub-claims, defined within NEDC-34140P (Reference 5-46)

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

In terms of the RCS reducing risks, a key risk that has informed the design of the BWRX-300 is large break LOCAs. This risk has been reduced by utilising redundant RIVs integral to the RPV, which eliminates unisolable LOCA faults. Two RIVs are installed in series on each major RPV line, with each being independently able to isolate the line and thus protect the RPV from fluid loss in the event of a pipe failure. The RIVs are designed to fail-safe, i.e., in the closed position, under fault conditions.

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

| L3 No. | Level 3 Chapter Claim: | Chapter 5 Arguments: | Sub-sections 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) | 5.3.3 RCS normal operating modes (5.3.3 & Section 16.6)<br>5.4.3 Core support structures<br>5.4.5 Chimney head & steam separator<br>5.4.6 Steam dryer<br>5.4.7 FW spargers<br>5.7.6 Leak detection I&C<br>5.8 Isolation Condenser System [ICS overpressure protection] (5.8 & 6.2.1)<br>5.9.2 RPV Pedestal<br>5.10.9 ICS RIVs |
| | | A record of safe BWR plant operation and continuous improvement demonstrates a well-founded design | NEDC-34137P, “BWRX-300 Design Evolution,” (Reference 5-47). |
| | | Safety functions associated with the relevant SSC have been substantiated during hazard and fault conditions | Safety function will be identified in PSR Ch. 3 and PSR Ch. 15. Means of substantiation are included in this chapter:<br>5.3.3 RCS normal operating modes (5.3.3 & Section 16.6)<br>5.4.3 Core support structures<br>5.4.5 Chimney head & steam separator<br>5.4.6 Steam dryer<br>5.4.7 FW spargers<br>5.7.6 Leak detection I&C<br>5.8 & 6.2.1 Isolation Condenser System [ICS overpressure protection]<br>5.9.2 RPV Pedestal<br>5.10.9 ICS RIVs |

| L3 No. | Level 3 Chapter Claim: | Chapter 5 Arguments: | Sub-sections 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 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 (Reference 5-47). OPEX & ALARP approach alignment which relate to the following sections:<br>• 5.1.1 RPV (Section 4.2),<br>• 5.1.2 & 5.12.4 Control Rods (Section 4.4),<br>• 5.2 & 5.10.1 RIVs (Section 4.3),<br>• 5.4.1 Reactor Internal (Section 4.2),<br>• 5.8 Isolation Condenser System (Section 4.5),<br>• 5.12.2 Reactor Water Clean-up System (Section 4.9). |
| | | The SSC have been designed in accordance with relevant codes and standards (RGP) | 006N3441 (Reference 5-32).<br>NEDC-34139P, “BWRX-300 UK Codes and Standards Assessment,” (Reference 5-48). |
| | | 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 BWRX-300 Safety Strategy, supplemented by the BWRX-300 General Description. These principles are also be presented within PSR Ch. 3:<br>006N5064, “BWRX-300 Safety Strategy” (Reference 5-49),<br>005N9751, “BWRX-300 General Description” (Reference 5-50). |
| 2.1.4 | System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning. | SSC pre-commissioning tests (e.g., 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. |

| L3 No. | Level 3 Chapter Claim: | Chapter 5 Arguments: | Sub-sections and/or reports that evidence the arguments: |
|---|---|---|---|
| | | 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 5-51) describes how safety categorisation and SSC classification are linked to quality group (QA arrangement) definition.<br>006N8706, “BWRX-300 Construction Strategy Report,” (Reference 5-52) describes the high-level construction quality assurance and quality control arrangements and responsibilities. |
| 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.<br>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:<br>006N6279, “BWRX-300 In Service Inspection Requirements” (Reference 5-53). |
| | | 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:<br>NEDC-34137P (Reference 5-47).<br>Degradation mechanisms for RPV and pressure boundary components is discussed in Section 5.2.4. |
| 2.1.6 | The BWRX will be designed so that it can be decommissioned safely, using current | 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:<br>[1] Materials are selected to minimise the quantities of radioactive waste and assisting decontamination, |

| L3 No. | Level 3 Chapter Claim: | Chapter 5 Arguments: | Sub-sections and/or reports that evidence the arguments: |
|---|---|---|---|
| | available technologies, and with minimal impact on the environment and people | | [2] Plant layout is designed to facilitate access for decommissioning or dismantling activities,<br>[3] Future potential requirements for storage of radioactive waste.<br><br>See NEDC-34193P, “BWRX-300 UK GDA Ch. 21: BWRX-300 Decommissioning and End of Life Aspects,” (Reference 5-54). |
| | | SSC are designed in order to minimise impacts on people and the environment during decommissioning. | See NEDC-34193P (Reference 5-54). |
| **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. | 006N3441 (Reference 5-32).<br>NEDC-34139P (Reference 5-48). |
| | | SSC have been designed in accordance with relevant codes and standards (RGP). | 006N3441 (Reference 5-32).<br>NEDC-34139P (Reference 5-48).<br>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. |
| 2.4.2 | 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 (Reference 5-47). |
| | | Any reasonably practicable design changes to reduce risk have been implemented | NEDC-34137P (Reference 5-47). |

| L3 No. | Level 3 Chapter Claim: | Chapter 5 Arguments: | Sub-sections and/or reports that evidence the arguments: |
|---|---|---|---|
| 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 5-55). |
| | | Design optioneering has considered all reasonably practicable measures. | See 006N3139 (Reference 5-55).<br>NEDC-34137P (Reference 5-47). |
| | | Any reasonably practicable design changes to reduce risk have been implemented. | NEDC-34137P (Reference 5-47). |

# 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 or Regulatory Observation

There are no FAP items identified associated with PSR Ch. 5.