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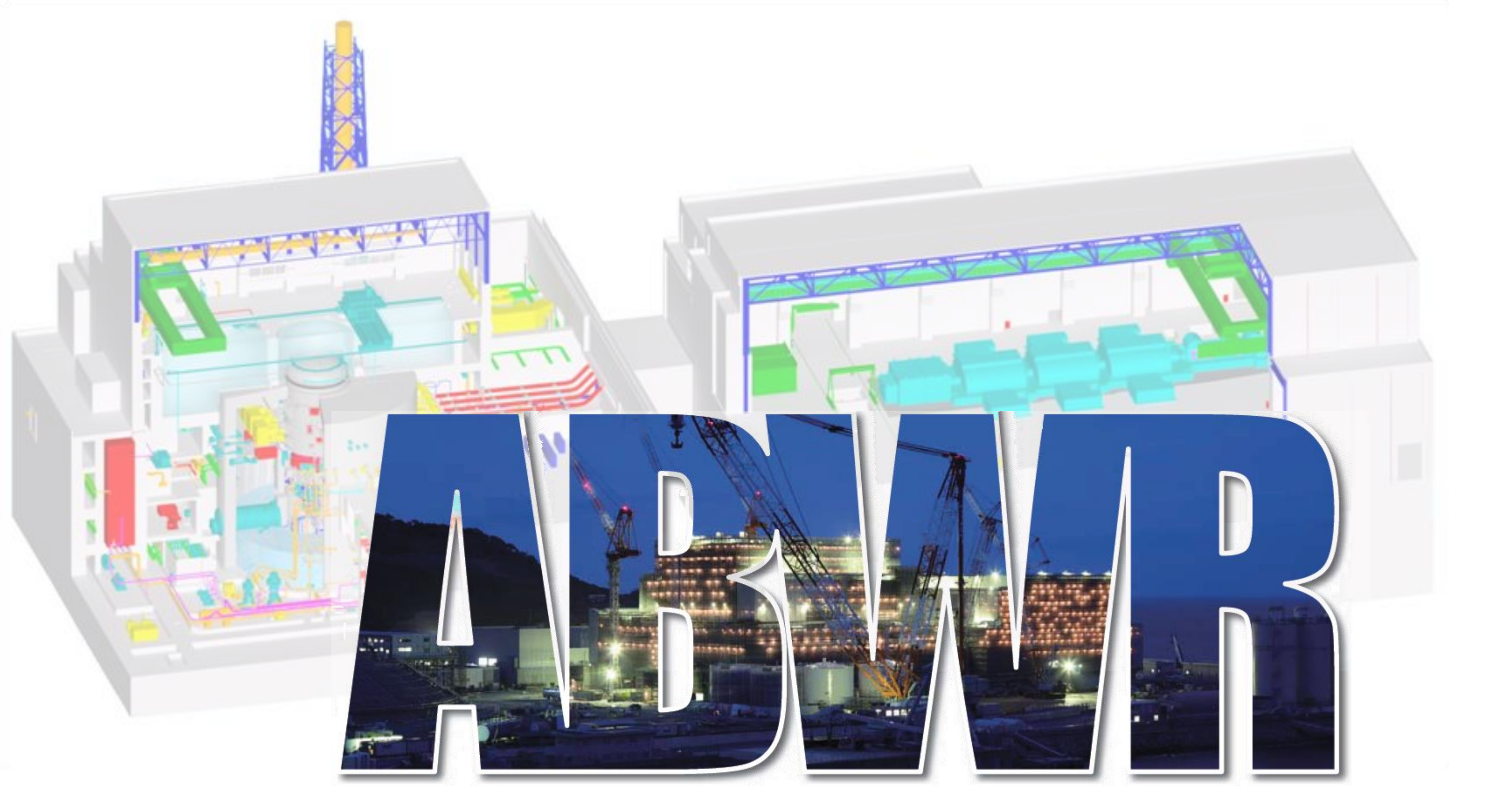
UK ABWR Generic Design Assessment

Generic PCSR Chapter 12 : Reactor Coolant Systems, Reactivity Control Systems and Associated Systems



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Executive Summary

This mechanical systems chapter describes the safety case for the UK ABWR Reactor Coolant Systems (RCSs), Reactivity Control Systems and Associated Systems. It lists the Safety Functional Claims (SFCs) that are made on the systems described in this chapter, together with the Safety Properties Claims (SPCs) that enable the process of demonstrating compliance of these systems with the Environmental Design Principles (NSEDPs).

The information provided includes: system design; functionality in normal conditions and during fault conditions; safety functional categorisation and system classification; important support systems; safety case assumptions, Limits and Conditions for Operation (LCO); and compliance with the As Low As Reasonably Practicable (ALARP) principle.

The overall Pre-Construction Safety Report (PCSR) justification that the UK ABWR is safe and satisfies the ALARP principle is underpinned by hazards assessments, design basis analysis, probabilistic safety analysis, beyond design basis analysis and human factors analysis (described in Chapters 6, 7 and 24 to 27), which demonstrate that the design of the systems covered by this chapter are fault tolerant. These analysis chapters specify the high level SFCs but do not specify requirements for design parameters on individual RCSs, Reactivity Control Systems and Associated Systems. Instead they apply analysis conditions and assumptions that are based on, and fully consistent with, the design information and safety claims for the systems that are presented in this chapter, in order to substantiate those claims.

The designs of all of the Safety Class 1 and 2 subsystems and components within the RCSs, the Reactivity Control Systems and the Associated Systems are well advanced for GDA, being largely based on proven technology from the Japanese ABWR reference design (hereinafter, J-ABWR). Additional risk reduction measures have been introduced (with reference to the J-ABWR design) in response to safety assessments undertaken in Generic Design Assessment (GDA), and these include an uprating of the performance of the Residual Heat Removal System (RHR) heat exchangers and enhancements to the Anticipated Transients Without Scram (ATWS) countermeasures to comply with UK expectations.
Other specific issues that have been considered in ALARP assessments include consideration of whether it is ALARP to retain the Bottom Drain Line and the Reactor Pressure Vessel Head Spray.
This chapter demonstrates that the risks associated with the design and operation of the RCSs, the Reactivity Control Systems and the Associated Systems for the UK ABWR are ALARP. It is acknowledged that further work will be required post-GDA to develop the design and fully incorporate site-specific aspects. This work will be the responsibility of any future licensee.

12.1 Introduction

This chapter of the PCSR describes the design features and various modes of operation of the RCSs, the Reactivity Control Systems and the Associated Systems for the UK ABWR. It lists all of the SFCs that are made on these systems to maintain the High Level Safety Functions (HLSFs) during normal and fault conditions as well as all of the SPCs, which are used in lower tier reference documents to support the demonstration that the SSCs described in this chapter comply with the NSEDPs (Ref 12.1-6).

12.1.1 Background

The main functions of the RCS are:

- To forcibly circulate the light water coolant/moderator through the core and remove heat generated by nuclear fission in the core, and
- to transport the high-temperature, high-pressure steam generated in the core to the turbine in order to drive it. After the steam has been used to drive the turbine, it is condensed into condensate, and this condensate is supplied back to the core.

The main functions of the reactivity control systems are:

- To control the reactivity of the core during normal operations including shutdown to maintain core conditions within design limits,
- to maintain adequate shutdown margin of the reactor when required,
- to rapidly bring the reactor to an emergency shutdown (Scram) if required, using control rods, and
- to provide a backup to the normal Scram system by providing a diverse means of inserting negative reactivity into the core using a liquid neutron absorber.

The main functions of the associated systems are:

- To provide overpressure protection,
- to provide detection of reactor coolant leakage,
- to recirculate a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities with their associated corrosion and fission products from the reactor coolant,
- to cool the nuclear system by removal of the decay heat,
- to provide water makeup to the core in the event of Loss of Coolant Accident (LOCA), and
- to collect leakage from the valve gland of some valves.

The design information provided in this chapter and its supporting documents is an important source of input data used in various assessments and analyses, reported in other PCSR chapters, that substantiates safety of operation of the UK ABWR. In particular the design basis analysis, probabilistic safety analysis, beyond design basis analysis and human factors analysis, described in Chapter 24: Design Basis Analysis to Chapter 27: Human Factors of the PCSR, demonstrate that the reactor and support systems are fault tolerant. The assumptions made in those analyses are fully consistent with the design information and safety claims for the RCSs, the Reactivity Control Systems and the Associated Systems that are presented in this chapter.

12.1.2 Document Structure of Chapter 12

The following sections are included in this chapter:

Section 12.2 Purpose and Scope
This section explains the objectives of the chapter, and lists the main systems that are within the scope of this chapter.

Section 12.3 Reactor Coolant Systems and Associated Systems
The RCS incorporates those systems and components that deliver steam to and convey condensate from the turbine and the condenser. The scope of Structures, Systems and Components (SSCs) of the RCS and associated systems that are described in this section is defined in sub-section 12.2.2 below.

Section 12.4 Reactivity Control Systems
The scope of reactivity control systems described in this section is defined in sub-section 12.2.2 below.

Sections 12.3 and 12.4 both include the following for the relevant SSCs:

- Design information including system configurations,
- normal operation and safety functions delivered, and the various modes of operation,
- safety functional categorisation and classification of the SSCs,
- safety claims made on the SSCs following the formal Claims-Arguments-Evidence (CAE) approach, and
- the required support systems to fulfil the safety case claims.

Section 12.5 Summary of ALARP Justification
This section provides a summary of the justification that the risks associated with the SSCs within Chapter 12 scope are acceptable (in terms of radiation dose consequences) and have been reduced to levels that are ALARP. This section refers to the results of the analyses/assessments in Chapter 24: Design Basis Analysis to Chapter 27: Human Factors that include representation of systems within Chapter 12 scope.

Section 12.6 Assumptions, Limits and Conditions for Operation
This section summarises the limits and conditions for operation that are specified in greater detail in the Basis of Safety Case (BSC) and Topic Report (TR) documents for the SSCs in the scope of this chapter. Assumptions are not covered in this chapter because there are no fundamental assumptions related to these SSCs.

Section 12.7 Conclusions
This section provides a summary of the main aspects of this chapter.

Section 12.8 References
This section lists all documents referenced within this chapter.

Other relevant information is captured in Appendices as follows:
Appendix A – Safety Functional Claims Tables
The claim trees for the SSCs in this chapter shown in Appendix A are a simplified version of the detailed claim trees contained in the BSC or TR of the related SSCs.

Appendix B – Safety Properties Claims Tables
The nine generic SPCs for all Mechanical Engineering (ME) SSCs that define the design requirements applicable to the SSCs scope of this chapter are presented in Appendix B tables as ME

SPCs. These tables of SPCs were derived for the ME SSCs based on the 'guide word' approach specified in Hitachi-GE's Safety Case Development Manual (SCDM) (Ref 12.1-5). Having derived the SPCs, a mapping exercise was undertaken to ensure that the SPCs fully cover the relevant NSEDPs applicable to the ME area. More information on the development of SPCs, and the coverage, at the more detailed level in the safety case, to demonstrate full compliance with the relevant NSDEPs is presented in Chapter 5.3 and the 'Topic Report on Safety Requirements for Mechanical SSCs' (Ref 12.1-3). Fulfilment of the requirements from the SPCs is justified in the BSC or TR of the related SSC as well as the 'Topic Report on Mechanical SSCs Architecture' (Ref 12.1-4).

Appendix C – Document Map
The document map showing Level 2 documents that support this chapter is provided in Appendix C.

The main links of this chapter with other PCSR chapters are as follows:

- For links to General Environmental Permit (GEP) and Conceptual Security Arrangements (CSA) documentation, please refer to Chapter 1: Introduction. For GEP, where specific references are required, for example in Radioactive Waste Management, Radiation Protection, Decommissioning, these are included in the specific sections within the PCSR.
- The general principles for the identification of Assumptions, Limits and Conditions for Operation (LCOs) related to the SSCs within this chapter scope are described in Chapter 4: Safety Management throughout Plant Lifecycle.
- The categorisation of safety functions and safety classification of the SSCs in this chapter conform with the methodology described in Chapter 5: General Design Aspects. The general requirements for equipment qualification, Examination, Maintenance, Inspection and Testing (EMIT) and codes and standards that come from this safety categorisation and classification are also described in Chapter 5. Further details can be found in the related sections of the BSC and TR documents of the SSCs within scope of this chapter.
- Hazard assessments (e.g. flooding, fire, rotating equipment related hazards, etc.) to demonstrate adequate performance of the SSCs within this chapter scope are included in Chapters 6: External Hazards and Chapter 7: Internal Hazards.
- Additional requirements for the SSCs that are classified as Very High Integrity (VHI), beyond those required for standard Class 1 components, are described in Chapter 8: Structural Integrity.
- The design of the reactor fuel and its support structures is done in detail in Chapter 11: Reactor Core.
- The design of additional safety functions of some reactor coolant systems such as the reactor core cooling function as part of the Emergency Core Cooling System (ECCS) of the Nuclear Boiler System (NB) and the Residual Heat Removal System (RHR) , or the containment heat removal function of the RHR, etc. are described in Chapter 13: Engineered Safety Features.
- The design of the SSCs scope of this chapter from the Control and Instrumentation point of view is described in detail in Chapter 14: Control and Instrumentation.
- The design of the SSCs scope of this chapter from the Electrical point of view is described in detail in Chapter 15: Electrical Power Supplies.
- The design of the mechanical systems supporting operation of the SSCs scope of this chapter such as cooling water supply, Heating Ventilating and Air Conditioning (HVAC), compressed air supply, etc. is described in detail in Chapter 16: Auxiliary Systems.
- The design of the reactor coolant systems beyond the Reactor Building (Main Steam System and Condensate and Feedwater System on the turbine side, etc.) are described in Chapter 17: Steam and Power Conversion Systems.
- Demonstration, using transient analysis, of the adequate performance of the SSCs within this chapter scope during design basis events and beyond design basis events is covered in

Chapters 24: Design Basis Analysis and Chapter 26: Beyond Design Basis and Severe Accident Analysis.
- Probabilistic analysis that demonstrates adequate reliability of the SSCs within this chapter scope is in Chapter 25: Probabilistic Safety Assessment.
- Substantiation of Human Based Safety Claims related to human interactions with the SSCs within this chapter scope is described in Chapter 27: Human Factors.
- An overview of how the UK ABWR design has evolved, and how this evolution contributes to the overall ALARP case is described in Chapter 28: ALARP Evaluation.
- Claims on the availability of the SSCs covered by this chapter to contribute to safe decommissioning operations and general requirements for decommissioning of these SSCs are described in Chapter 31: Decommissioning.

This chapter is supported by a set of reference documents, primarily BSCs and TRs (Level 2 documents) and their associated Level 3 documents, including System Design Descriptions (SDDs). Each BSC or TR describes a specific system or component within the scope of Chapter 12, explaining where the arguments and evidence that substantiate the safety claims for those SSCs are presented. A full list of the Level 2 documents is provided within the document map in Appendix C.

12.2 Purpose and Scope

12.2.1 Purpose

The overall purpose of this chapter is to provide a definitive source of design information for all aspects of the UK ABWR RCSs, Reactivity Control Systems and Associated Systems that have an impact on nuclear safety during normal and fault conditions. All claims made on those systems in the UK ABWR safety case are identified and substantiated as far as is possible with the current design maturity at the conclusion of the GDA.

Specific objectives of the chapter are:

(1) Identify and describe the SSCs within the scope of the RCSs, the Reactivity Control Systems and Associated Systems.
- The physical attributes of the relevant SSCs, including such things as simple system line diagrams, design limits, required minimum plant performance characteristics (e.g. pump flow rates), etc.
- The functions delivered, and the various modes of operation of those SSCs which have multiple functions/modes of operation.

(2) Identify and describe the safety functions of the SSCs within the scope of the chapter, and to specify the safety functional categorisation of those functions.

(3) Specify the safety classification of the SSCs within the scope of the chapter, which includes some SSCs that are classified as VHI.

(4) Specify all relevant safety case claims, and describe or provide pointers to where the detailed sub-claims, arguments and evidence for each claim can be found in the supporting BSCs, TRs, and the detailed Level 3 design information.
- These are SFCs and SPCs. Apart from some system SFCs described as ‘top claims’, each SFC and SPC has a unique identifier that conforms to the conventions defined in the SCDM (Ref 12.1-5).

(5) Identify the support systems required for all the SSCs within the chapter scope, and make reference to where the arguments and evidence that substantiate all relevant safety claims on these are presented in supporting documents or other chapters of the GDA PCSR.
- Control and Instrumentation systems, Electrical Power Supplies, and Water Supplies, if relevant.

(6) Identify the main links to relevant content of other GDA PCSR chapters, to ensure consistency across the whole safety case, and to ensure the overall safety case presented in the GDA PCSR is complete.

(7) Provide or identify references to relevant evidence that demonstrates that the risks associated with failure of the SSCs within the scope of this chapter are ALARP.

(8) Identify aspects of the design substantiation of SSCs within the scope of this chapter that require further work beyond completion of GDA.

12.2.2 Scope

The scope of the systems included in this chapter is as follows:

Description of the RCS and Associated Systems:

- Reactor Pressure Vessel (RPV), which contains the reactor core,
- Reactor internal structures (excluding the fuel assemblies and control rods which are described in Chapter 11: Reactor Core),
- Reactor Coolant Pressure Boundary (RCPB), and its Overpressure Protection System that uses Safety Relief Valves (SRVs),
- Leakage Detection System (LDS),
- Reactor Recirculation System (RRS), including the Reactor Internal Pumps (RIPs),
- NB, which is divided into the Main Steam System (MS) and the Feedwater System (FDW) (the Main Steam Isolation Valves (MSIVs) are part of the NB),
- Reactor Water Clean-up System (CUW),
- RHR,
- Valves and Valve Gland Leakage Treatment System (VGL), and
- Component Supports.

Description of Reactivity Control Systems:

- Control Rod Drive System (CRD), and
- Standby Liquid Control System (SLC).

This chapter describes all modes of the RHR, but further details of the ECCS functions of the RHR are included in Chapter 13: Engineered Safety Features.
This chapter also presents the Reactor Core Isolation Cooling System (RCIC) and the High Pressure Core Flooder System (HPCF) as part of the RCPB, but they are fully described in Chapter 13: Engineered Safety Features as part of the ECCS.

12.3 Reactor Coolant Systems and Associated Systems

12.3.1 Summary Description of the Reactor Coolant Systems and Associated Systems

12.3.1.1 Reactor

The UK ABWR is a Boiling Water Reactor of the internal steam-water separation type with recirculation pumps built inside the reactor vessel for a forced-circulation direct cycle using light water as the coolant and moderator. The rated thermal power of this reactor is 3,926MW, but the main safety facilities of the reactor (the SRVs, the ECCS, the Reinforced Concrete Containment Vessel (RCCV), etc.) are conservatively designed for a thermal power of 4,005MW (equivalent to the 102% rated power conditions assumed in the safety analysis).

The basic components of the reactor include the RPV, the internal structures of the RPV, the core, the Control Rods (CRs) and their drive mechanisms. Figure 12.3-1 and Figure 12.3-2 illustrate the relative arrangement of these components. The RPV contains built-in components such as the core support structures, the RIPs, the feedwater and emergency core cooling water distribution headers, the Steam Separator and the Steam Dryer.

The core is surrounded by the Core Shroud. The Core Shroud isolates the upward flow of coolant inside the core from the downward flow of coolant in the annulus between the Core Shroud and the RPV wall. This annulus is described as the ‘downcomer annulus’.

The Core Shroud is a stainless-steel cylinder and is supported by the Shroud Support. Since there are no nozzles connecting to large diameter pipes below the top of the core, even in the event of a pipe break, the core will remain covered by the coolant because the ECCS will operate to compensate for the coolant flow escaping through the largest pipe break.

The fuel assemblies making up the core are grouped in sets of four. Each group is supported by the orificed fuel support located on the top of the Control Rod Guide Tubes (CRGTs). Each individual CRGT is supported by a CRD housing, which is welded to the bottom of the RPV.

The fuel assemblies where there are no CRGTs are supported by the Peripheral Fuel Support located on the Core Plate, and the Core Plate is supported by the Core Shroud.

There is a Top Guide for supporting the top of the fuel assemblies in the lateral direction. The Top Guide is also supported by the Core Shroud.

The CRs are inserted into and withdrawn from the core through CRGTs at the lower plenum of the reactor. Each CR is connected to a Fine Motion Control Rod Drive (FMCRD) by a mechanical coupling.

Ten RIPs are arranged at the bottom of the RPV, equally-spaced in the annular region between the Core Shroud and RPV. The motors of the RIPs are installed into the motor casings, which are mounted at the bottom of the RPV. The RIP motors drive impellers inside the RIP diffusers by shafts. The RIP diffusers penetrate through the Shroud Support Plate so that reactor coolant flow is forced through the diffusers by the RIP impellers into the lower plenum.

The flow distribution of the coolant through the RPV is shown in Figure 12.3-3. The rates of coolant flow going to the peripheral fuel and the central fuel assemblies are adjusted to the required proportions by the orifices located inside the Fuel Supports shown in Figure 12.3-4, to ensure an adequate coolant flow is conducted to all of the fuel assemblies. Coolant is heated while passing through the fuel assemblies until boiling occurs and it becomes a two-phase flow of steam and water. The two-phase flow leaving the core enters the shroud head. The shroud head is for mixing evenly the two-phase flow coming separately from each fuel assembly before it enters the Standpipes of the Steam Separator.

The two-phase flow brought into the Steam Separator units is separated into steam and water by the action of centrifugal forces. The steam exits the Separator and enters into the Steam Dryer, where nearly all of the remaining moisture in the steam is finally removed, and the dry steam then leaves the vessel through four Main Steam (MS) outlet nozzles on the RPV.

The water separated from the steam in the Steam Separator and the Steam Dryer flows down through the downcomer annulus between the Core Shroud and the RPV wall, mixes together with the feedwater entering the RPV through the Feedwater Spargers, and re-circulates through the RIPs and the reactor core again.

The core wide power changes of the reactor during operation are usually controlled by adjusting the coolant recirculation flow rate. The reactivity control system positions the CRs to provide fine tuning of the core power distribution. If the reactor cannot be brought sub-critical during normal operation due to an inability to insert the CRs, the reactor is brought sub-critical to cold shutdown conditions by means of the SLC, which injects sodium pentaborate solution with sufficient negative reactivity worth to achieve cold shutdown.

In order to decrease the corrosion products in the coolant and the radioactivity associated with them, the parts coming into contact with the coolant are made of suitable materials with considerations for corrosion resistance as described in Chapter 8: Structural Integrity.

Table 12.3-1 gives the main design specifications of the RPV and the core.

12.3.1.2 Reactor Internal Structures

Overview

The reactor internal structures of the RPV consist of the core support structures which support the core and the other internal structures.

The core support structures include the Core Shroud, Shroud Support, Top Guide, Core Plate, Fuel Support and CRGTs. The other internal structures include the Steam Separator, Shroud Head, Steam Dryer, RIPs, Feedwater Spargers, HPCF coupling and spargers, Low Pressure Flooder (LPFL) spargers, Differential Pressure Line, and Head Vent and Spray Nozzles, etc.

The core support structures support and position the core in order to keep the structural integrity of Fuel Assemblies, regarding the appropriate load combinations (during normal operation, in the event of abnormal operational transients, and in the event of accidents).

The reactor internal structures also separate water from the steam to the appropriate steam quality requirements and form the flow path of coolant to appropriately cool the Fuel Assemblies during normal operation and in the event of abnormal operational transients.

Main Components

(1) Core-Support Structures

(a) Core Shroud

The Core Shroud is a cylinder made of stainless-steel. It has a structure which forms the coolant flow path ascending through the core, while at the same time separating the ascending flow from there re-circulation flow descending in the downcomer annulus outside the Core Shroud. The lower part of the Core Shroud is supported by the Shroud Support which is welded to the RPV bottom.

(b) Top Guide and Core Plate

The Top Guide is machined into a grid shape from a stainless-steel disk. It is fastened onto the top of the Core Shroud and serves to support and guide the fuel assemblies in the horizontal direction. It also supports the tops of the in-core flux monitors and the start-up neutron sources.

The Core Plate is a stainless steel disk reinforced by beams. It is fastened into the lower portion of the Core Shroud and provides lateral support to the CRGTs, the Fuel Supports and fuel assemblies, the In-Core Guide Tubes and the start-up neutron sources.

(c) Control Rod Guide Tubes

The CRGTs serve to guide the CRs. They extend upwards from the top of the CRD housings and fit into the Core Plate. The load of the fuel assemblies is transmitted to the RPV through the Orificed Fuel Supports, the Core Plate and Core Shroud and the CRGTs.

(2) Reactor Internals

(a) Steam Separator

The Steam Separator is of the vertical, axial-flow centrifugal type. It consists of 349 Steam Separator units arranged in parallel over the Shroud Head. Figure 12.3-5 illustrates the configuration of the Steam Separator units. On exiting from the fuel assembly within the core, the steam-water mixture enters the core top plenum to allow mixing to occur before entering the Steam Separators. The steam-water mixture enters the bottom end of the Steam Separator units, through the Standpipes. Here a rotational motion is given to the fluid by the inlet vanes. Centrifugal effects separate water and steam, while the mixture is moving upwards in the tube, by free vortex movements. The water is carried up the inside surfaces of the Steam Separator units and gathered at the top of the tube and is channelled out of the Steam Separator units. Meanwhile, the steam proceeds vertically upwards in the direction of the axis, and enters the Steam Dryer.

(b) Shroud Head

The Shroud Head supports the Steam Separators, and creates the plenum for steam-water mixture. The Shroud Head also channels separated water back to the downcomer annulus where it is re-circulated. The Shroud Head is removed during each outage, in conjunction with the Steam Separators and the Shroud Head Bolts.

(c) Steam Dryer

The elements within the steam dryer, as shown in Figure 12.3-6, are made from parallel corrugated plates. While the steam passes between the corrugated plates, its direction changes a number of times. Each time the direction changes, the moisture droplets in the steam collide with the surface of the corrugations, and moisture is removed. Then the dry steam leaves the RPV through the MS outlet nozzles and is conducted to the turbine. The water separated by the steam dryer is drained out of the steam-dryer via the drain troughs and returned to the upper side of the shroud head for recirculation.

(d) Reactor Internal Pumps

Ten RIPs are arranged at the bottom of the RPV and are equally-spaced in the annular region between the core shroud and RPV. The saturated water coming from the steam separators and steam dryer above the core is mixed with the feedwater, pressurised by the RIPs, and supplied to the core via the diffusers of the RIPs.

The impellers and diffusers of the RIPs are inside the RPV, as shown in Figure 12.3-7. The impellers are connected to the motors via shafts, and the diffusers are fastened onto the RIP nozzles (discussed in detail in section 12.3.5.1, Reactor Recirculation System).

(e) Feedwater Sparger

The Feedwater Sparger is installed to distribute the feedwater that enters the RPV through the feedwater lines and nozzles. The Feedwater Sparger enables the sub-cooling of the recirculation flow by mixing together in a uniform manner the feedwater with high-temperature coolant separated in the Steam Separator and Steam Dryer.

The Feedwater Spargers consist of six, independent sparger headers (with elbows), which diverge to the right and left in a Tee along the RPV inner wall. Each sparger header is supported with pins to the Feedwater Sparger Bracket. The six headers each connect to the six Feed Water nozzles and feedwater lines. Each sparger has a number of distribution nozzles to ensure appropriate distribution.

(f) LPFL Sparger

The LPFL Sparger is a part of the ECCS described in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System, the LPFL spargers are provided for cooling the core suitably in an accident. The LPFL spargers are located within the downcomer annulus between the Core Shroud and the RPV wall just below the Feedwater Spargers.

There are two LPFL Spargers, each independent headers (including elbow nozzles), and supported by a bracket in the same manner as the Feedwater Spargers. The LPFL Sparger has a series of nozzles to distribute the flow evenly and to ensure no possibility of blockage.

(g) HPCF Coupling and HPCF Sparger

Of the ECCS described in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System, two sets of the HPCF Couplings and the HPCF Spargers are part of the HPCF and are provided for cooling the core at high pressure following an accident. The HPCF Coupling, which enters from the two HPCF nozzles of the RPV, is connected to the HPCF Spargers which are mounted within the Top Guide directly above the reactor core. The HPCF Sparger has a series of nozzles to distribute the flow evenly over the core and to ensure no possibility of blockage.

(h) Head Vent and Spray Nozzle

The Head Vent and Spray Nozzle are provided for spraying CUW water from the top of the RPV in order to cool the top of the RPV when the reactor is shut down. The Head Vent maintains a small flow through the RPV head vent line to one of the MS lines during operation, to ensure no incondensable gases build up in the RPV head.

(i) Differential Pressure Line

The Differential Pressure Line consists of the RIP Differential Pressure Line and the Core Plate Differential Pressure Line. These are used for core flow measurement, by way of measuring the differential pressure across the Core Plate and the RIP.

Four pairs of the RIP Differential Pressure Line enter the RPV by penetrating the RPV bottom head as a concentric double pipe. The inner and outer pipes of the RIP Differential Pressure Line become independent pipes within the RPV, with one pipe open to the suction side and one to the discharge side of the RIP. Four pairs of the Core Plate Differential Pressure Line similarly penetrate the RPV bottom head as a concentric double pipe and opens to the bottom and top of the Core Plate. Core flow measurement by the RIP Differential Pressure Line is used for core performance calculation, normal operation supervision and calibration of core flow measurement by means of the Core Plate differential pressure. Core flow measurement by Core Plate Differential Pressure Line is monitored by the reactor protection system, and is also used for recirculation flow control, etc.

(j) In-Core Guide Tubes

The In-Core Guide Tubes are welded to the top of the In-Core Housings, which are connected to the bottom head of the RPV. They are laterally supported by the Core Plate and guide the in-core flux monitors.

(k) Fuel Channels

The fuel channels perform the function to form the fuel bundle flow path outer periphery for bundle coolant flow, where a fuel bundle consists of fuel rods, water rods, expansion springs, spacers, and upper and lower tie plates. The fuel channels also perform the function of providing surfaces for CR guidance in the reactor core, and to provide structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers (discussed in detail in Chapter 11: Reactor Core).

(l) Start-up Range Neutron Monitor Drytube

The Start-up Range Neutron Monitor (SRNM) Drytubes are installed within the core through the RPV bottom, In-Core Housing and In-Core Guide tube in order to contain and protect the SRNM detectors. SRNM is a part of Neutron Monitoring System (NMS). For further information of NMS, see Chapter 14: Control and Instrumentation, Section 14.7.3.

Table 12.3-2 gives the main specifications of the reactor internal structures, and Figure 12.3-1and Figure 12.3-2 illustrate the structures inside the RPV.

12.3.1.3 Reactor Coolant Systems and Associated Systems

Overview

The RCS consists of the systems that deliver steam to and convey condensate from the Turbine and the Condenser. The RCS consists of the following SSCs:

- The RPV,
- the RRS,
- the NB, which is divided into the MS and the FDW within the Reactor Building (R/B), and the MS and Condensate and Feedwater System (CFDW) beyond the R/B up to and including the Turbine Building (T/B), and
- the MS turbine and the Main Condenser.

Figure 12.3-8 gives a schematic diagram of the RCS, and shows how it interfaces with the Steam and Power Conversion System.

The RPV and a portion of the RRS, the MS and the FDW form a major part of the RCPB. Figure 12.3-9 gives an outline of the RCPB.

The MS on the turbine side, the CFDW, the turbine and the condenser are described in Chapter 17: Steam and Power Conversion Systems.

Particularly, the RCSs have the following functions:

(1) They forcibly circulate the light water coolant/moderator through the core and remove heat generated by nuclear fission in the core.

(2) They transport the high-temperature, high-pressure steam generated in the core to the turbine in order to drive the turbine. After the steam has been used to drive the turbine, it is condensed into condensate, and this condensate is supplied back to the core.

Summary Description

As mentioned above, part of the RCS forms a major portion of the RCPB. This sub-section provides information regarding the RCS and the associated systems connected to it including their pressure-containing components up to and including isolation valves at the RCPB boundary. This grouping of components is defined as the RCPB.

The RCPB, illustrated in Figure 12.3-9, includes pressure-retaining components such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the RCS,

or

(2) connected to RCS up to and including any and all of the following:
    - The outboard containment isolation valve in piping which penetrates the Primary Containment Vessel (PCV) (e.g. MS pipework).
    - The second of the two valves normally closed during normal reactor operation in system piping which does not penetrate the PCV.
    - The RCS SRV piping.

The following is a summary description of the RCSs and the main Associated Systems connected to them that form part of the RCPB.

(1) MS

The MS accepts steam from the RPV. The steam flows through the RPV MS outlet nozzle flow restrictors, the MS lines, and the MSIVs.

The MS Flow Restrictors are venturi-type and are included in each MS nozzle on the RPV inside the PCV. The restrictors are designed to limit the rate of loss of coolant following a MS line break inside or outside the PCV. The coolant loss is limited so that the reactor vessel water level remains above the top of the core during the time required for the MSIVs to close. This action protects the fuel barrier to release of radioactive material.

The restrictors also limit the maximum pressure differences expected across the reactor internal components following complete severance of a MS line.

Two MSIVs are installed on each MS line. One is located inside and the other outside the PCV. If a MS line break occurs inside the containment, closure of the isolation valve outside the PCV seals the containment itself. The MSIVs automatically isolate the RCPB when a pipe break occurs outside the containment. This action limits the loss of coolant and the potential for release of radioactive materials from the nuclear system.

The MS protects the RCPB from damage due to overpressure by providing pressure-operated relief valves that can discharge steam from the nuclear system to the Suppression Pool (S/P). The MS also acts to automatically depressurise the nuclear system in the event of a LOCA in which the FDW, the RCIC and the HPCF fail to maintain RPV water level. Depressurisation of the nuclear system allows the low pressure flooder systems to supply enough cooling water to adequately cool the fuel.

(2) FDW

The portion of the FDW, contained in the R/B, provides a piping path for feedwater flow, CUW return flow, and RHR and RCIC flows into the RPV. The continuation of the FDW beyond the R/B belongs to the CFDW of the Turbine Building and is described in Chapter 17: Steam and Power Conversion Systems, Section 17.10 Condensate and Feedwater System.

(3) RRS

The RRS provides coolant flow through the core. Adjustment of the core coolant flow rate changes reactor power output. The RIPs are located inside the RPV, thus eliminating large piping connections to the vessel below the core and also eliminating the external recirculation piping.

(4) RHR

The RHR includes a number of pumps and heat exchangers that can be used to cool the nuclear system under a variety of situations. During normal shutdown and reactor servicing, the RHR removes decay heat. The RHR also allows decay heat to be removed whenever the main heat sink (Main Condenser) is not available. Some modes of the RHR operation allow the removal of heat from the PCV following faults such as LOCA. Another operational mode of the RHR is the LPFL mode which allows injection of water to cool the reactor at low pressure following an accident. In addition, the RHR can be used to provide Spent Fuel Storage Pool water cooling under certain circumstances.

The LPFL is an engineered safety feature for use during a postulated LOCA. Operation of the LPFL is presented in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System.

(5) CUW

The CUW recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities with their associated corrosion and fission products from the reactor coolant. It also controls the inventory of coolant in the reactor system under controlled conditions (e.g. during start-up or shutdown).

(6) RCIC

The RCIC provides makeup water to the core during events in which feedwater flow is not available such as faults or for reactor shutdown. The system is started automatically upon receipt of a low reactor water level signal, high drywell pressure signal, or manually by the operator. Water is pumped to the core by a turbine-driven pump using driving steam from the reactor steam lines.

(7) HPCF

The HPCF provides makeup water to the RPV to assure that sufficient reactor inventory is maintained in order to perform adequate core cooling and prevent reactor fuel overheating during transient (as a backup of the RCIC) and accident conditions. Like the RCIC, the system is started automatically upon receipt of a low reactor water level signal, high drywell pressure signal or manually by the operator.

Table 12.3-1 : Main Design Specifications of the Reactor

Reactor thermal power	3,926 MW
Reactor Internal Pump	10 units
Coolant full flow rate	approx. 52.2×10^3 t/h (rated core flow)
Core inlet sub-cooling	approx. 54.2 kJ/kg
Average steam mass flow rate at core outlet	approx. 14.5% by mass
Reactor pressure (RPV dome part)	approx. 7.07 MPa [gauge]
Steam flow rate	approx. 7.64×10^3 t/h
Steam pressure	approx. 7.07 MPa [gauge]
Steam temperature	approx. 287°C

Table 12.3-2: Main Design Specifications of the Reactor Internal Structures

(1) Core Shroud	
Number	1
Type	Cylinder type
Material	Stainless steel
(2) Top Guide	
Number	1
Type	Grid shape
Material	Stainless steel
(3) Core Plate	
Number	1
Type	Disk type
Material	Stainless steel
(4) Control Rod Guide Tubes	
Number	205
Type	Cylinder type
Material	Stainless steel
(5) Steam Separator Units	
Number	349
Type	Vertical, axial-flow centrifugal type
Material	Stainless steel
Flow rate	
Steam	Approx. 22 t/h/unit
Water	Approx. 144 t/h/unit
Carry-under (Steam Separator outlets)	Less than 0.25 wt%
(6) Steam Dryer	
Number	1
Type	Corrugated plate
Material	Stainless steel
Moisture (Steam -Dryer outlets)	Less than 0.1 wt%
(7) Reactor Internal Pump	
Number	10
Capacity	Approx. 7,700 m^3/h/unit

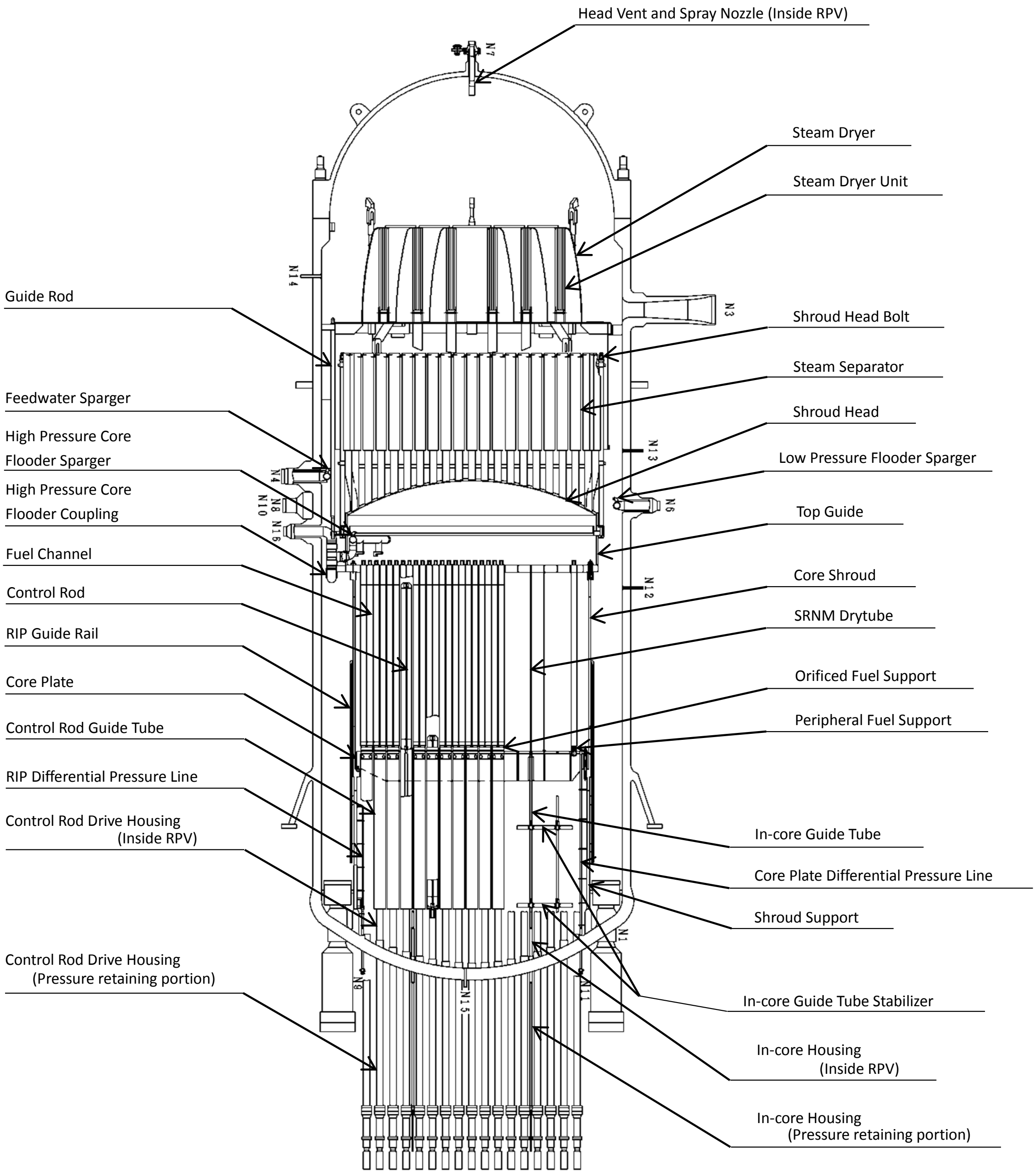

Figure 12.3-1 : Reactor Internals and RPV (cross section)

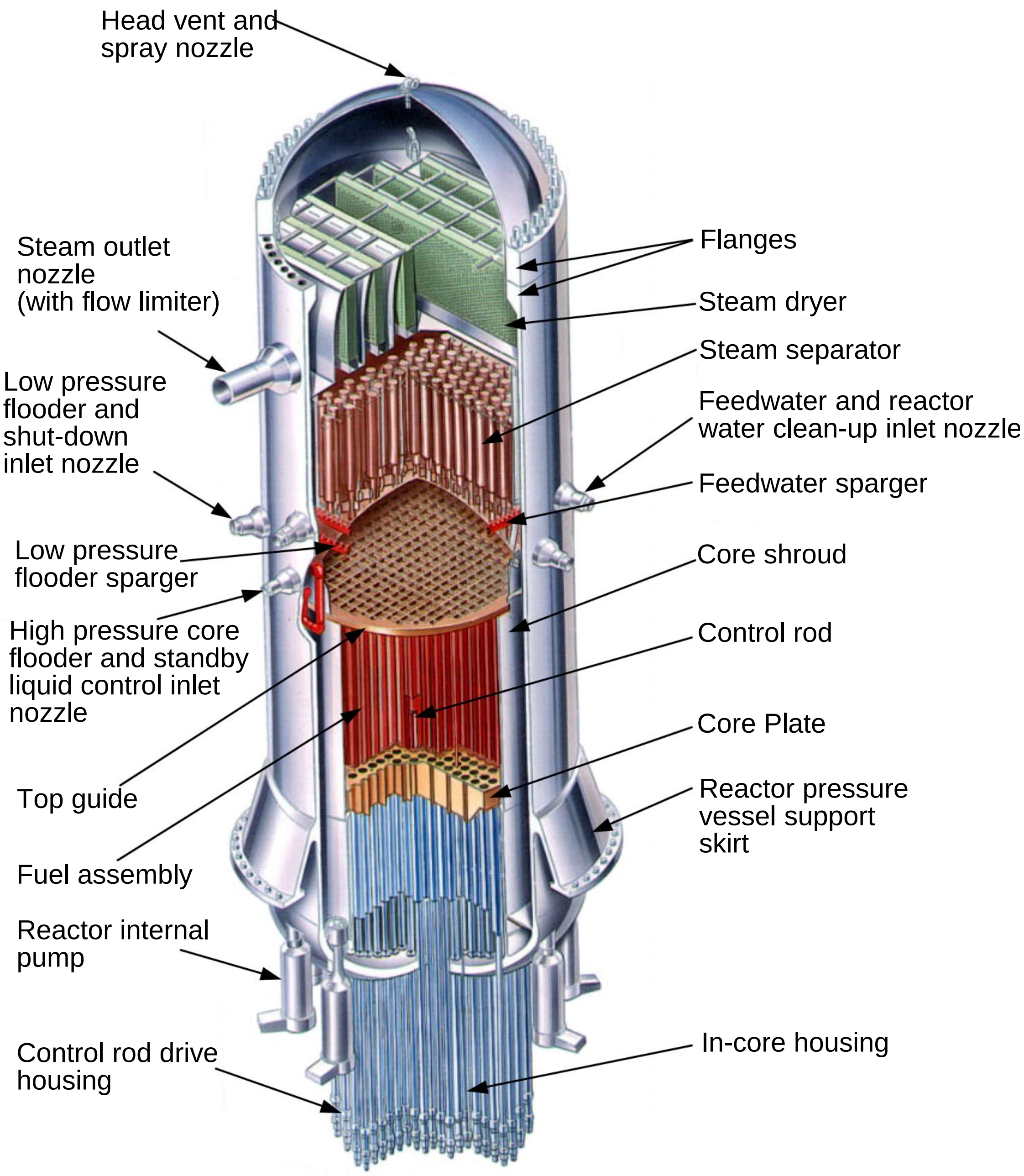

Figure 12.3-2 : Reactor Internals and RPV (sketch)

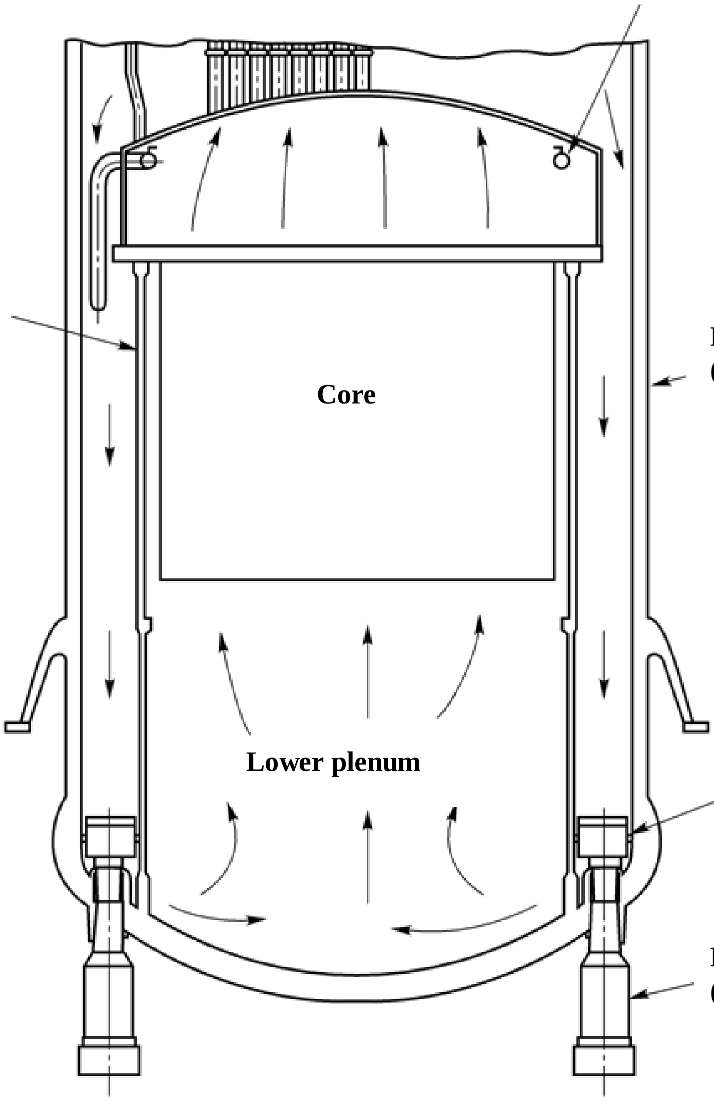

Figure 12.3-3 : Coolant Flow Paths inside the RPV

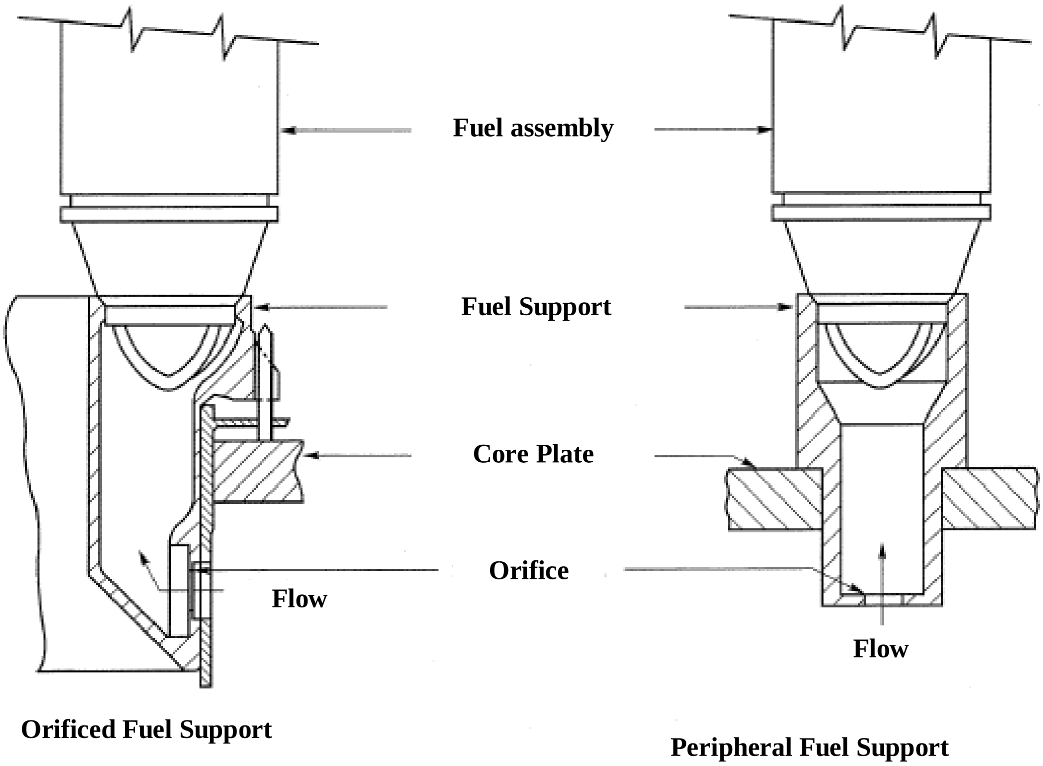

Figure 12.3-4 : Outline Drawing of the Fuel Supports

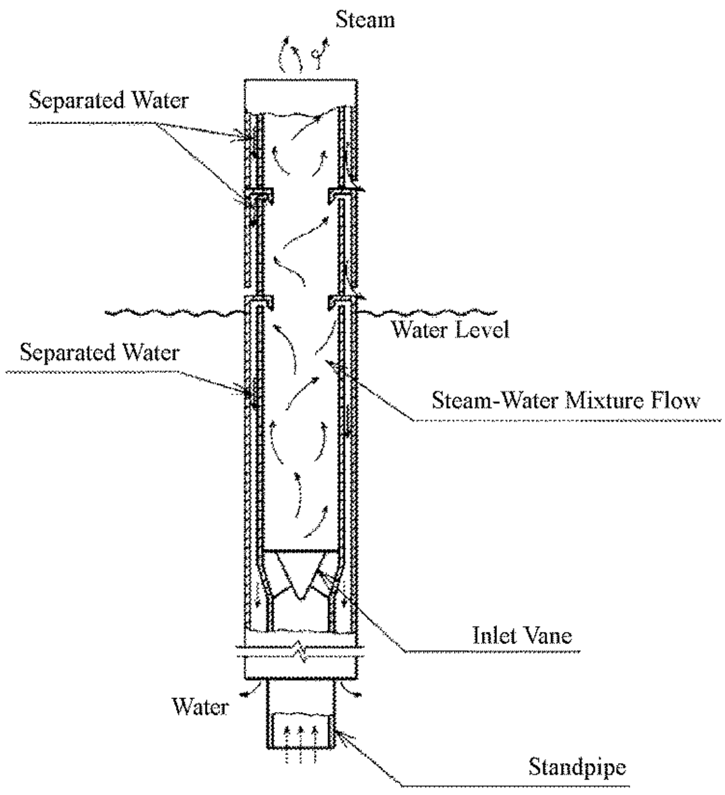

Figure 12.3-5 : Outline Drawing of a Steam Separator Unit

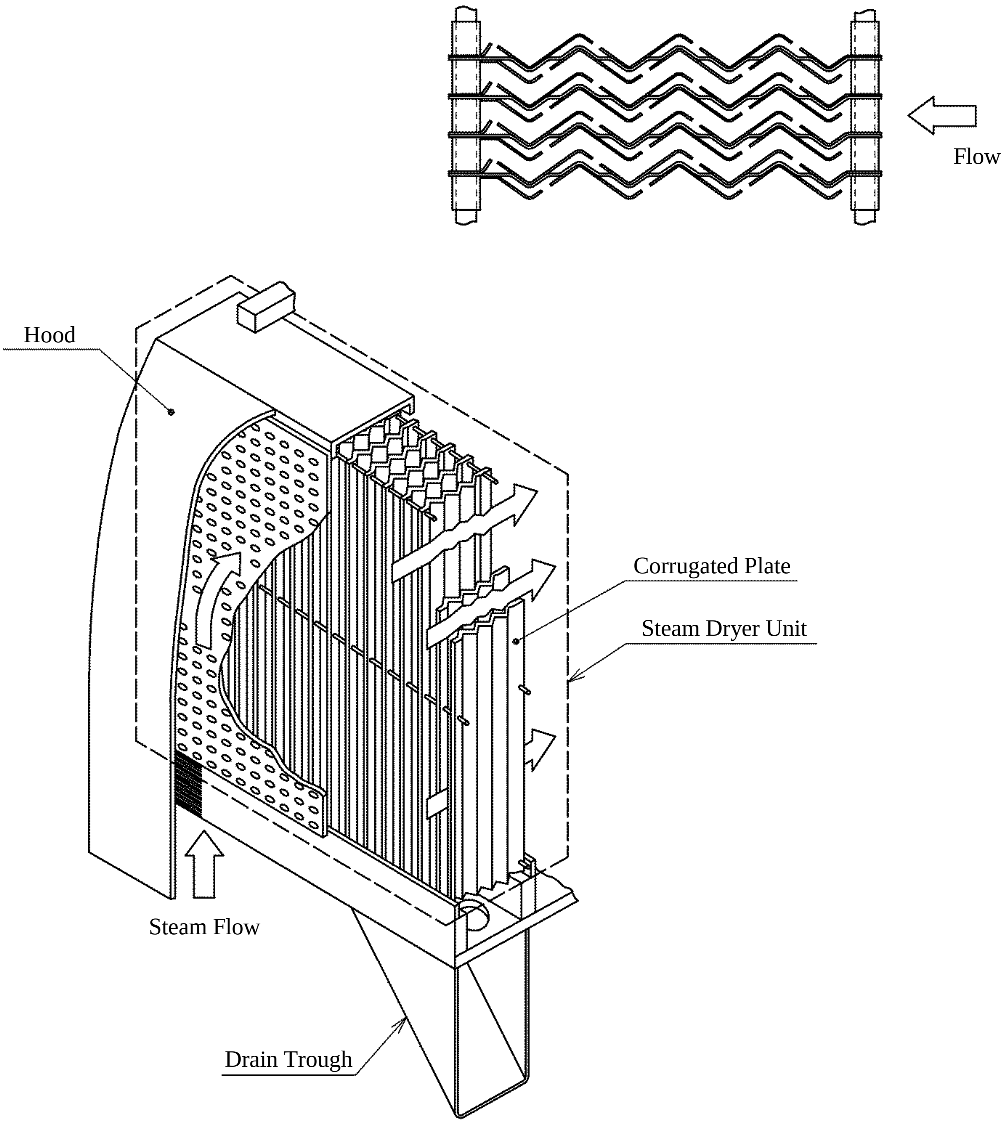

Figure 12.3-6 : Outline Drawing of the Steam Dryer

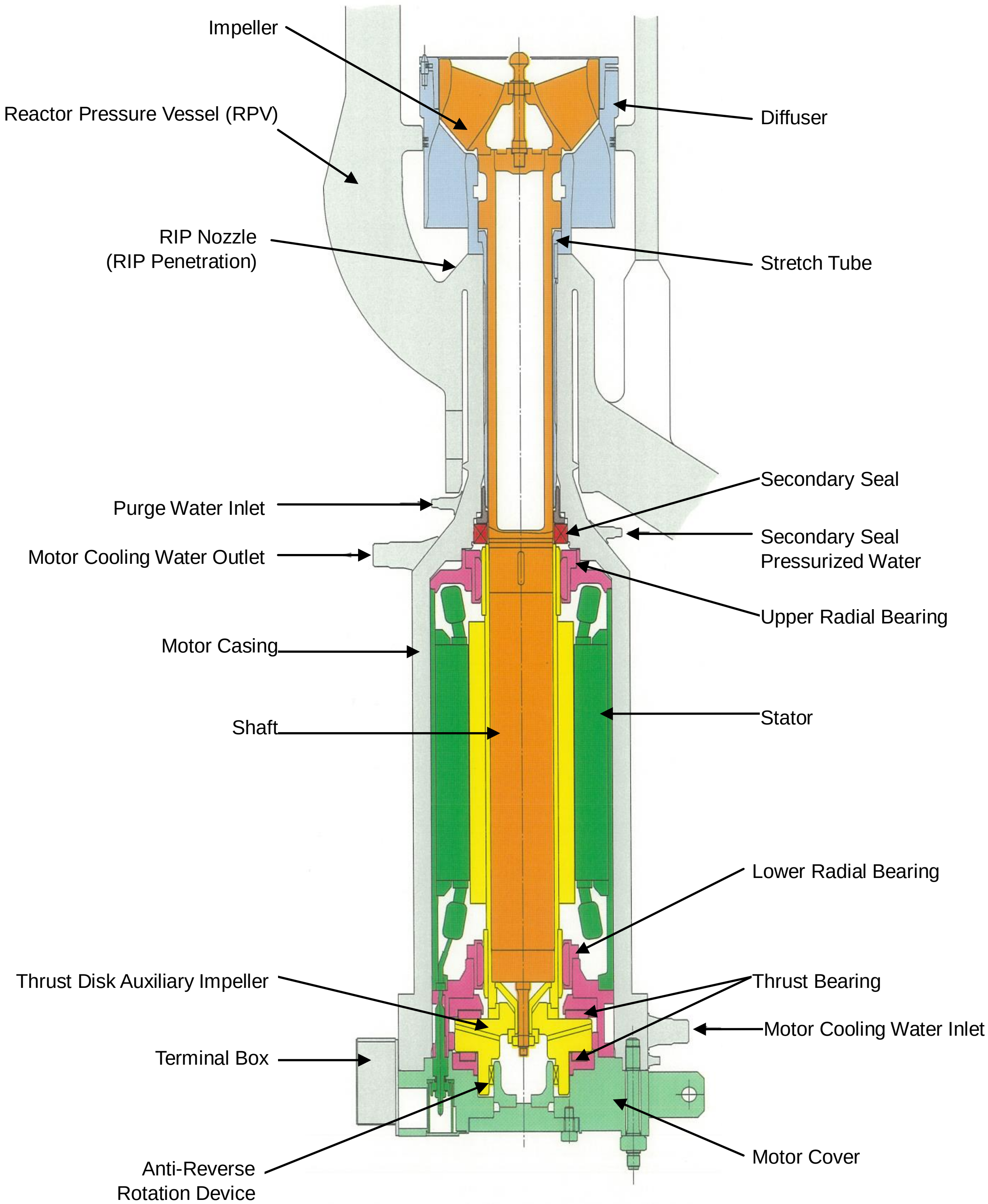

Figure 12.3-7 : Outline Drawing of the Reactor Internal Pump

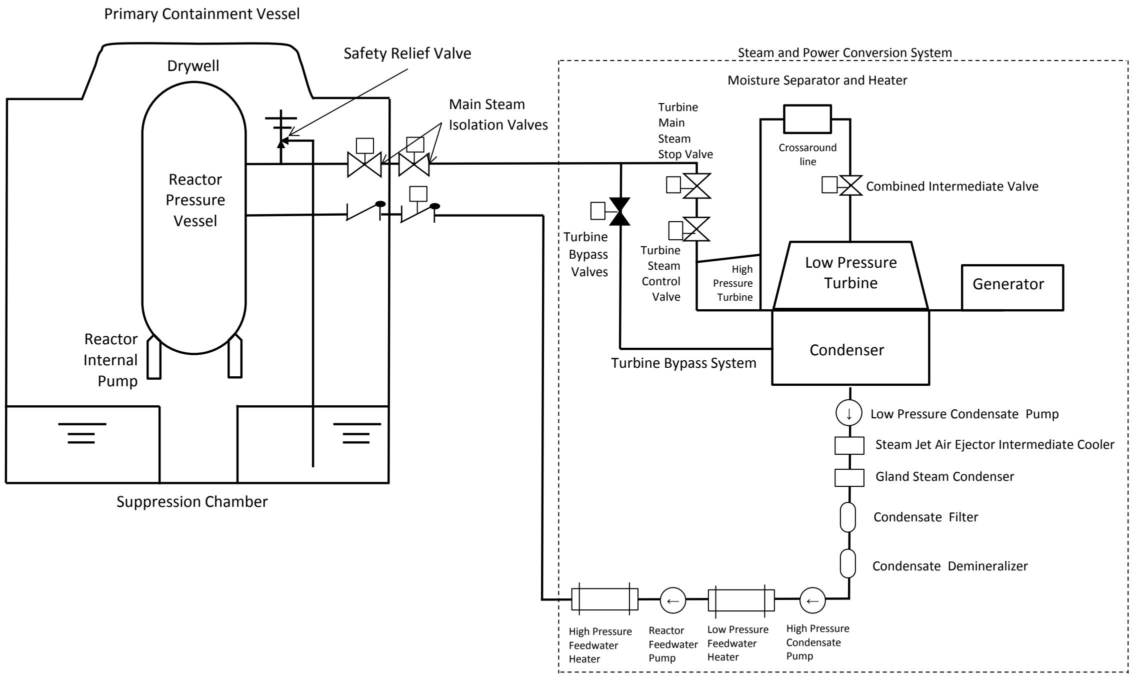

Figure 12.3-8 : Schematic Diagram of the Reactor Coolant System

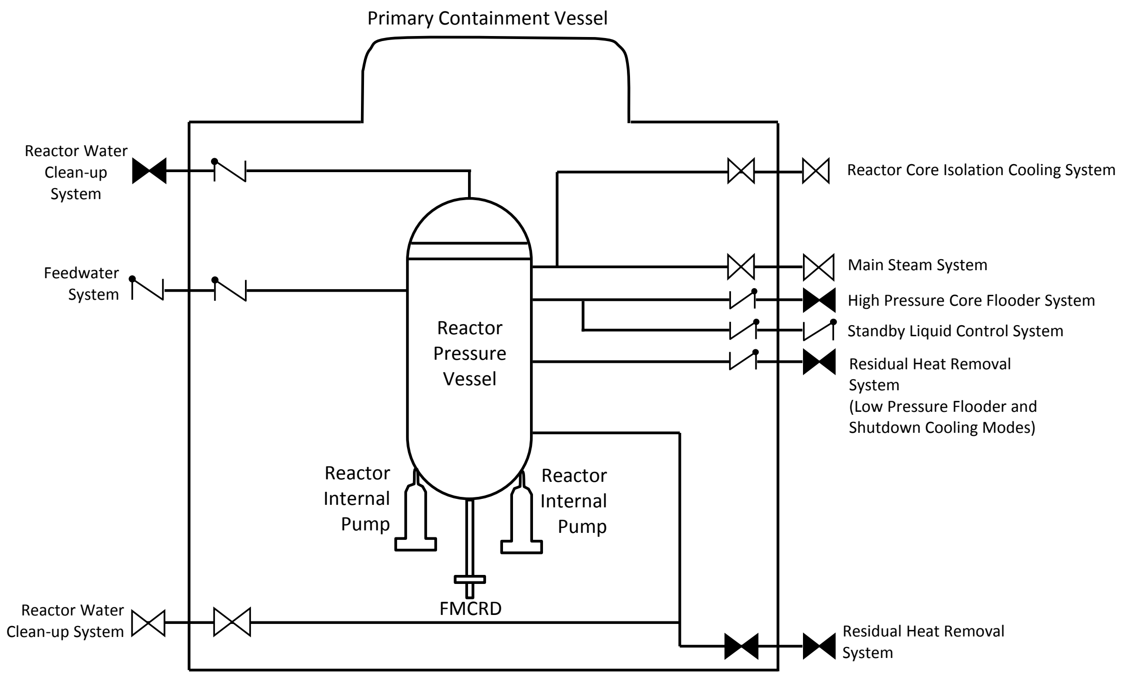

Figure 12.3-9 : Outline Drawing of the RCPB

12.3.2 Safety Requirements

The safety requirements for the RCSs and Associated Systems are described in the system design bases of the corresponding sub-section for each SSC, respectively.

12.3.3 Reactor Coolant Pressure Boundary (RCPB)

12.3.3.1 Design Bases

Structural Integrity

The components constituting the RCPB are designed to assure the required structural integrity during all operating conditions. Loss of the structural integrity would lead to consequences above the Basic Safety Level (BSL) as defined in Chapter 5: General Design Aspects, Section 5.5. From this perspective, the RCPB delivers a Category A safety function as defined in Chapter 5: General Design Aspects, Section 5.6.3.

This general claim is divided into the following sub-claims, which correspond to the components forming the RCPB:

(1) The RRS portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [RRS SFC 4-1.1]

(2) The NB portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [NB SFC 4-1.1]

(3) The CUW portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [CUW SFC 4-1.1]

(4) The RHR portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [RHR SFC 4-1.1]

(5) The RCIC portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [RCIC SFC 4-1.1]

(6) The HPCF portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [HPCF SFC 4-1.1]

(7) The CRD portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [CRD SFC 4-1.1]

(8) The SLC portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [SLC SFC 4-1.1]

The components necessary to deliver these functions are classified as VHI, High Integrity (HI) or Class 1 safety components according to the Structural Integrity classification of the UK ABWR. The RCPB function is ensured by providing each one of the SSCs forming it with the reliability commensurate with their Structural Integrity classification. This is achieved by compliance with the requirements of well-established and appropriate design codes. This approach is described in Chapter 8: Structural Integrity as indicated in Section 12.3.3.2.

(1) The RRS SSCs forming part of the RCPB are designed with reliability commensurate with Standard Class 1 in order to ensure [RRS SFC 4-1.1].
(2) The NB SSCs forming part of the RCPB are designed with reliability commensurate with VHI or Standard Class 1 in order to ensure [RRS SFC 4-1.1].
(3) The CUW SSCs forming part of the RCPB are designed with reliability commensurate with Standard Class 1 in order to ensure [CUW SFC 4-1.1].
(4) The RHR SSCs forming part of the RCPB are designed with reliability commensurate with Standard Class 1 in order to ensure [RHR SFC 4-1.1].
(5) The RCIC SSCs forming part of the RCPB are designed with reliability commensurate with Standard Class 1 in order to ensure [RCIC SFC 4-1.1].
(6) The HPCF SSCs forming part of the RCPB are designed with reliability commensurate with Standard Class 1 in order to ensure [HPCF SFC 4-1.1].
(7) The CRD SSCs forming part of the RCPB are designed with reliability commensurate with Standard Class 1 in order to ensure [CRD SFC 4-1.1].
(8) The SLC SSCs forming part of the RCPB are designed with reliability commensurate with Standard Class 1 in order to ensure [SLC SFC 4-1.1].

Overpressure Protection

The RCPB is designed to provide the pressure relief functions in order to prevent its overpressurisation in design basis faults that could otherwise lead to failure. Loss of the overpressure protecting function would lead to consequences above the BSL. From this perspective, it includes safety valves which deliver a Category A safety function and the components necessary to deliver this function are classified as Class 1 safety components according to the safety categorisation and classification of UK ABWR (defined in Chapter 5: General Design Aspects, Section 5.6).

This general claim is divided into the following sub-claims, which correspond to the system delivering overpressure protection:

(1) The MS through the safety valve function of the SRVs is the principal means to deliver overpressure protection of the RCPB under abnormal transients and accident conditions that could put excessive pressure on the boundary. [NB SFC 4-2.1]
(2) The MS is a principal means to prevent excessive loss of reactor coolant after SRV actuation for the safety valve function. [NB SFC 4-5.1]
(3) The MS through the relief valve function of the SRVs is an additional means to deliver overpressure protection of the RCPB faster than the safety valve function under abnormal transients and accident conditions that could put excessive pressure on the boundary. [NB SFC 4-6.1]

These safety functions are developed and justified in section 12.3.3.3 about the RCPB Overpressure Protection System.

Isolation

Piping connected to the RPV and the reactor coolant system, and part of which constitutes the RCPB are designed to include isolation valves in order to prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the RPV and in order to shut off the release of radioactive materials external to the PCV resulting from a design basis fault such as a piping break outside the PCV. Loss of the isolating function would lead to consequences above the BSL. From this perspective, the isolation valves deliver a Category A safety function and the components necessary to deliver this function are classified as Class 1 safety components according to the safety categorisation and classification of UK ABWR (defined in Chapter 5: General Design Aspects, Section 5.6).

This general claim is developed and justified in detail in Chapter 13: Engineered Safety Features, Section 13.3.3.2 Primary Containment Isolation System.

Leakage Detection

The RCPB is designed with features to detect leakage from it during normal operation and design basis faults in order to initiate rapid closure of isolation valves. Loss of the leakage detecting function would potentially lead to consequences above the BSL. From this perspective, the leakage detection from the RCPB delivers a Category A safety function and the components necessary to deliver this function are classified as Class 1 safety components according to the safety categorisation and classification of UK ABWR (defined in Chapter 5: General Design Aspects, Section 5.6).

This general claim is divided into the following sub-claims, which correspond to the system delivering leakage detection:

(1) The LDS is a principal means to deliver monitoring of leakage within the reactor coolant system. [LDS SFC 4-3.1]

(2) The LDS is a principal means to deliver detection and alarm of leakage from the reactor coolant system as well as initiation of the signals to isolate the corresponding systems in the event a leakage is detected. [LDS SFC 4-7.1]

These safety functions are developed and justified in section 12.3.3.4 about the RCPB Leakage Detection System.

12.3.3.2 Structural Integrity

In the UK, it is a requirement of the regulatory regime for a Licensee to provide a safety case to demonstrate how the structural integrity of nuclear power plant SSCs can be assured over the design life of the plant to a level of structural reliability and a degree of rigour commensurate with the consequences of postulated gross failure. In order to achieve this, a key principle of the strategy is that the level of rigour applied to the structural integrity justification is related to the nuclear safety consequences of failure. The component classification methodology provides a structured approach for justifying structural integrity of SSCs commensurate with the consequences of their failure.

The safety functional requirements of SSCs are identified from the UK ABWR fault schedule. The UK ABWR categorisation of safety functions and classification of components are determined by the approach described in Chapter 5: General Design Aspects, Section 5.6 Categorisation and Classification of Structures, Systems and Components (SSCs). Three broad classifications are defined; these are Class 1 to 3.

The frequency and consequences of failure of Class 1 components will vary significantly. As the risk of failure varies, so will the required assurance of structural integrity. In order to identify where the very highest standards of structural integrity should apply, Ref 12.3-24 describes a refined scheme of classification to be adopted which sub-divides Class 1 into three; VHI, HI and Standard Class 1.

The detailed method of classification of VHI and HI components is identified and summarised in Chapter 8: Structural Integrity.

Structural integrity classification is allocated based on the components safety function, the direct consequences of failure, the lines of protection and the indirect consequences of failure. Classification of VHI and HI components, in some cases, is based on the Failure Modes and Effects Criticality Analysis (FMECA) technique from the candidate of the VHI and HI components. Details on the classification method are described in Ref 12.3-24.

Safety classification is being undertaken as an iterative process. For certain components, a classification has been assigned to expedite production of the topic reports that consider structural integrity. Parts of the RPV, some of the MS piping inside the PCV and inboard MSIVs are assigned classification of VHI. Reactor Internals, outboard MSIVs, some Feedwater piping and some MS piping have classification of Standard Class 1. Classification is completed by consideration of the results of fault analyses and reference to the single failure criterion.

Table 12.3-3 summarises the results of the classification of VHI and HI components.

The reliabilities that are required for VHI and HI components are achieved by application of special requirements beyond those required by normal nuclear codes applied for Standard Class 1 components, as described in Chapter 8: Structural Integrity.

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Table 12.3-3 : Structural Integrity Safety Classification of SSCs

Component	Weld Location / Region	Classification
RPV	RPV Major Boundary (Cylindrical Shell, Top Head / Bottom Head / Nozzles as a part of Vessel)	Very High Integrity
	・Small diameter Nozzles (i.e. Instrumentation nozzles or pipes / Drain nozzle) ・RIP Penetration ・CRD Penetration (i.e. CRD Housing) ・In-core Penetration (i.e. In-core Housing) ・Main Steam Outlet Nozzle Extension	Standard Class1
RPV Support Skirt	Each connecting weld	Standard Class1
RPV Stabilizer	Each connecting weld and Rod	Standard Class1
Core Support Structure	Top Guide / Core Plate / Shroud Support / Core Shroud / CR Guide Tube / Fuel Support	Standard Class1
MS	Each connecting weld inside containment vessel including Main Steam Outlet Nozzle Extension to piping weld / Inboard MSIV and MS RCCV penetration	Very High Integrity
	Each connecting weld outside containment vessel and MS nozzle to Main Steam Outlet Nozzle Extension weld / Outboard MSIV	Standard Class1
SRV	Bonnet bolt	Standard Class1
FDW	Each connecting weld / Valves / RCCV Penetrations	Standard Class1
SLC	Each connecting weld / Valves / RCCV Penetrations	Standard Class1
HPCF	Each connecting weld / Valves / RCCV Penetrations	Standard Class1
RHR	Each connecting weld / Valves / RCCV Penetrations	Standard Class1
RCIC	Each connecting weld / Valves / RCCV Penetrations	Standard Class1
CUW	Each connecting weld / Valves / RCCV Penetrations	Standard Class1
	Head Vent and Spray Nozzle (boundary portion)	Standard Class1
FMCRD (boundary portion)	Spool Piece	Standard Class1

Note that no SSCs have currently been assigned HI safety classification, but the possibility of assigning SSCs to the HI classification in the future is not foreclosed in GDA

12.3.3.3 RCPB Overpressure Protection System

System Summary Description

This section is a general introduction to the RCPB Overpressure Protection System where the system roles, system functions, system configuration and modes of operation are briefly described. The Overpressure Protection System safety case is justified in the 'Basis of Safety Cases on Reactor Coolant Pressure Boundary Overpressure Protection System' (Ref 12.3-1) and the 'Basis of Safety Cases on Nuclear Boiler System' (Ref 12.3-4). The system design is described in detail in the system design specifications (Ref 12.3-8) and the Piping and Instrumentation Diagrams (P&IDs) (Ref 12.3-9 to Ref 12.3-11).

(1) System Roles

The main role of the RCPB Overpressure Protection System is to protect the components forming the RCPB against overpressure generated during transient or accidents in order to ensure their integrity to preserve reactor coolant.

(2) Functions Delivered

The 16 SRVs which constitute the RCPB Overpressure Protection System are designed to perform the following functions:

(a) Safety valve function: the valves function as safety valves which open to prevent nuclear system overpressure. They are self-actuated by inlet steam pressure acting against a closing spring force if the relief valve function (described below) failed or was not effective enough to reduce the pressure inside the RCPB. In addition, the valves are provided with a re-seating function to prevent excessive depressurisation which could lead to an excessive loss of reactor coolant during safety valve operation.

(b) Relief valve function: the valves are opened using a pneumatic actuator upon receipt of an automatic or manually-initiated signal to reduce pressure or to limit a pressure rise.

(c) Simultaneously with depressurisation for RCPB overpressure protection by the safety valve function or the relief valve function, the SRVs also support the delivery of reactor core cooling by high pressure core cooling systems as described in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System.

(3) Basic Configuration

The RCPB Overpressure Protection System consists of 16 SRVs located on the MS lines between the RPV and the first MSIV within the drywell (D/W) of the PCV. The main components are listed below. The configuration of the SRVs on the MS lines is shown on Figure 12.3-10.

(a) SRV,

(b) SRV accumulators,

(c) SRV Discharge Line Quenchers, and

(d) valves, piping, instrumentation, and controllers.

Design Bases

This section describes the design bases for the RCPB Overpressure Protection System. The RCPB Overpressure Protection System has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

Fault Conditions

(1) The MS through the safety valve function of the SRV is the principal means to release the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV. [NB SFC 2-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(2) The MS through the relief valve function of the SRV is an additional means to the release of the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV. [NB SFC 2-1.2]

(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

(3) The MS through the safety valve function of the SRVs is the principal means to deliver overpressure protection of the RCPB under abnormal transients and accident conditions that could put excessive pressure on the boundary. [NB SFC 4-2.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(4) The MS is a principal means to prevent excessive loss of reactor coolant after SRV actuation for the safety valve function. [NB SFC 4-5.1]

(This function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements)

(5) The MS through the relief valve function of the SRVs is an additional means to deliver overpressure protection of the RCPB faster than the safety valve function under abnormal transients and accident conditions that could put excessive pressure on the boundary. [NB SFC 4-6.1]

(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

System Design Description

This section describes the design of the RCPB Overpressure Protection System to support and justify the delivery of [NB SFC 2-1.1], [NB SFC 2-1.2], [NB SFC 4-2.1], [NB SFC 4-5.1] and [NB SFC 4-6.1]. Additional design description can be found in Ref 12.3-1, Ref 12.3-8, and Ref 12.3-9 to Ref 12.3-12.

(1) Overall System Design and Operation

(a) Safety Valve Operation Mode

The NB is provided with 16 SRVs, all of which incorporate the safety valve mode of operation. The safety mode of operation (actuation by steam pressure) is initiated when the direct and increasing inlet steam pressure overcomes the restraining spring and the frictional forces (which will act against the inlet steam flow once established) at the main disk and causes the main disk to move to the open position. The condition at which this action is initiated is called the “setting lift pressure” and corresponds to the set-pressure value stamped on the nameplate of the SRV. Therefore, the safety valve function is passive and does not require any power or control system or any operator’s action. The safety valve function finishes when the steam pressure drops to the valve re-seat pressure, which is overcome by the spring force that closes the valve.

Based on the safety analysis presented in Chapter 24: Design Basis Analysis, the set pressure and the discharge capacity of the SRVs are strategically set at different values as shown on the following Table 12.3-4 after considering a margin in order to contain the RCPB pressure below 120% of the design pressure in the event of infrequent faults. The spring load establishes the safety valve opening set pressure and the valves are set to open sequentially at the setting lift pressures.

Table 12.3-4: Overpressure Safety Valve Operation Set Pressures

	Setting Lift Pressure (MPa [gauge])	Number of Valves	Capacity/Valve (t/h) (at 103% of setting lift pressure)
Safety Operation	approx. 7.92	2	approx.460
	approx.7.99	4	approx.464
	approx.8.06	4	approx.468
	approx.8.13	3	approx.472
	approx.8.20	3	approx.476

The SRVs main disk opening stroke times are within the times required by the safety analysis in order to achieve overpressure protection when they are spring-actuated.

Hence, the SRVs through the safety mode of operation passively open and close to depressurise the reactor in order to prevent over-pressurisation of the boundary during fault scenarios. For those scenarios where the reactor is cooled by the high pressure core cooling systems of the ECCS, the SRVs operate the same way to discharge the steam generated by the decay heat into the S/P as necessary in order to prevent excessive pressures that could impair operation of the high pressure core cooling systems while the reactor is being cooled.

(b) Safety Valve Re-seat Function

Every time the SRVs open, the steam discharged into the S/P represents a loss of reactor coolant. In order to prevent an excessive depressurisation which could lead to an excessive loss of reactor coolant during safety valve operation, the SRVs automatically re-seat when the steam pressure drops to the valve re-seat pressure, which is overcome by the spring force that closes the valve.

The SRVs are designed to re-seat as required by the safety analysis in order to prevent excessive loss of reactor coolant when the SRV is actuated.

(c) Relief Valve Operation Mode

The NB is provided with 16 SRVs, all of which incorporate the relief valve mode of operation as a defence in depth measure. The relief mode of operation (power actuation mode) is initiated when an electrical signal is received at the solenoid valve exclusive for this function located on the pneumatic actuator assembly. The solenoid valve will open, allowing pressurised nitrogen gas to enter the lower side of the pneumatic cylinder piston which pushes the piston and the rod upwards. This action pulls the lifting mechanism of the main disk, thereby opening the valve to allow inlet steam to discharge through the SRV until the solenoid valve closes again when the re-seat pressure is reached to cut off pressurised nitrogen gas to the actuator. Therefore, relief valve function is automatic and does not require operator’s action. The pneumatic actuator is so arranged that, if it malfunctions, it will not prevent the valve from opening when steam inlet pressure reaches the spring lift set pressure for safety valve operation mode.

Based on the safety analysis the set pressure and the discharge capacity of the SRVs are strategically set at different values as shown on the following Table 12.3-5 after considering a margin in order to contain the RCPB pressure below 110% of the design pressure in the event of frequent faults. For overpressure relief operation, the valves are provided with pressure-sensing devices which cause them to operate sequentially at the set points designated in this table.

Table 12.3-5: Overpressure Relief Valve Operation Set Pressures

	Setpoints (MPa [gauge])	Number of Valves	Capacity/Valve (t/h) (at 103% of setpoint pressure)
Relief Operation	approx.7.51	1	approx.422
	approx.7.58	1	approx.426
	approx.7.65	4	approx.431
	approx.7.72	4	approx.434
	approx.7.79	3	approx.438
	approx.7.86	3	approx.442

For each valve the relief operation setpoint is at a lower pressure than the safety operation setpoint.

The SRVs are connected to the High Pressure Nitrogen Gas Supply System (HPIN) which constantly supplies them with nitrogen gas from the nitrogen gas supply machine at sufficient pressure to actuate and open them. All the SRVs are provided with one three-way solenoid valve, which automatically opens when the set points are reached in order to open the nitrogen gas supply path to the actuation assembly. The solenoid valve can also be remotely and independently operated from the Main Control Room (MCR) to control the flow of nitrogen gas to the actuator of the SRVs.

All the SRVs are also provided with one nitrogen accumulator of 20L of capacity dedicated for relief valve function, which is kept charged by the HPIN nitrogen gas supply. Loss of the HPIN supply does not prevent opening of the SRVs because of the nitrogen gas stored in the accumulators. The capacity of the accumulator is sufficient to open and keep open the target SRVs under the pressure conditions in the PCV and RPV when the delivery of this function is required.

(2) Equipment Design and Operation

SRV

The SRVs passively open by spring actuation in order to deliver overpressure protection function in the safety valve operation mode.

The SRVs automatically open by powered actuation in order to deliver overpressure protection function in the relief valve operation mode.

The SRVs passively close when the steam inlet pressure reaches the re-seat set pressure in order to prevent excessive loss of reactor coolant when operating as safety valves by spring actuation.

For further details on the SRV design for the delivery of these functions refer to section 12.3.5.2' Nuclear Boiler System (NB)', where the whole design of the SRV is described.

(3) Main Support Systems

(a) Instrumentation and Control Systems

Instrumentation and control are provided to measure and control the operational conditions of the Overpressure Protection System components necessary for the delivery of the safety functions claimed. The status, measurements and alarms of the components and valves to be remotely operated are generally displayed in the MCR. The relevant instrumentation and control provisions are described as follows.

(i) Safety Valve Operation Mode
The main parameters measured in relation to the delivery of this function are the position of the SRVs and the temperature of the SRV discharge line, which are all displayed in the MCR Human-machine Interface (HMI).

(ii) Safety Valve Re-seat Function
The main parameters measured in relation to the delivery of this function are the position of the SRVs, which is displayed in the MCR HMI.

(iii) Relief Valve Operation Mode

- The SRVs are provided with pressure-sensing devices which cause them to automatically open at the set points designated in Table 12.3-5. Operation is controlled by the logic of the Other C&I System (OCIS).
- Operation of the SRVs for the relief valve mode is controlled from the MCR HMI, which also allows remote manual operation.
- The main parameters measured in relation to the delivery of this function are the position of the SRVs, the temperature of the SRV discharge line and the nitrogen supply pressure to the valves, which are all displayed in the MCR HMI.

(b) Power Supply System
   (i) Safety Valve Operation Mode
   The spring-actuated configuration is passive and therefore it does not require power supply.
   (ii) Safety Valve Re-seat Function
   The configuration is passive and therefore it does not require power supply.
   (iii) Relief Valve Operation Mode
   The three-way solenoid valve for actuation of the SRVs through the relief accumulator is supplied DC power by the Safety Class 3 Power Supply System.

Assumptions, Limits and Conditions for Operation

In order to ensure that the RCPB Overpressure Protection System is operated within the safety limits and that the design requirements from the safety case are met during the operating regime, appropriate LCOs and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the ‘Basis on Safety Cases on Reactor Coolant Pressure Boundary Overpressure Protection System’ (Ref 12.3-1).

- All the sixteen SRVs shall be operable during start-up, power operation and hot shutdown for the delivery of the SFCs claimed when required.

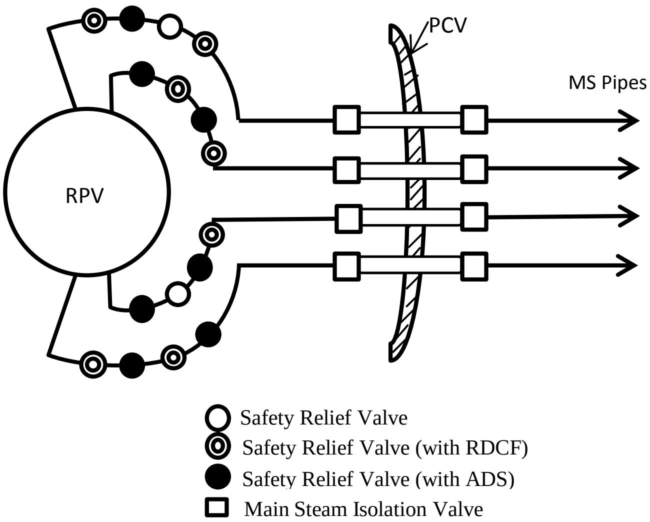

Figure 12.3-10 : Configuration of the SRVs

Notes:
The Automatic Depressurisation System (ADS) shown on this figure is explained in sub-section 12.3.5.2 and in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System.
The Reactor Depressurisation Control Facility (RDCF) shown on this figure is explained in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System and Chapter 16: Auxiliary Systems, Section 16.7.3.3 Reactor Depressurisation Control Facility.

12.3.3.4 Reactor Coolant Pressure Boundary and Core Cooling Systems Leakage Detection

System Summary Description

The RCPB and core cooling systems leakage detection is a primary function of the LDS. This section is a general introduction to the LDS. The LDS safety case is justified in the Basis of Safety Cases on Reactor Coolant Pressure Boundary and Core Cooling Systems Leakage Detection System (Ref 12.3-2). The LDS is described in detail in the system design specifications (Ref 12.3-13) and the P&IDs (Ref 12.3-14 and Ref 12.3-15).

(1) System Roles

The main role of the LDS is to detect leakage from the reactor coolant systems indicated below and initiate an alarm if the quantity of leakage exceeds the permissible limits. Moreover, the LDS is used to initiate the signals to isolate the systems concerned depending on the leakage when necessary.

(a) Reactor coolant systems located within the PCV.

(b) Reactor coolant systems located outside the PCV:

- NB (MS and FDW),
- CUW,
- RHR, and
- RCIC.

(2) Functions Delivered

(a) The LDS detects and alarms leakage from each system by monitoring a combination of the following parameters:

(i) Reactor water level.
(ii) Pressure and temperature in the PCV.
(iii) Water level of liquid waste sumps in the D/W.
(iv) Condensed water flow from the Drywell Cooling System (DWC) Cooling Unit.
(v) Airborne particulate activity in the D/W.
(vi) Flow rate in process piping .
(vii) Temperature in the areas where components and piping are installed.

(b) Small leakage within the D/W is detected by monitoring changes in pressures and temperatures, flow of drains into the liquid waste sumps, waste sump pumps initiation frequency and increases in the radioactivity concentration within the atmosphere inside the PCV.

(c) In the case of large leakage, leakage is detected by reactor water level reductions and changes in the process piping flow as well as the specified detections above.

(d) The LDS is used to initiate the signals to isolate the systems forming part of the PCV boundary depending on the leakage.

(3) Basic Configuration

The LDS consists of a group of instrumentation and controllers that belong to the LDS itself and also other systems that are necessary to detect a leakage of reactor coolant and send the appropriate alarm and isolation signals to the systems concerned. The main metering devices used are thermometers, manometers, flow meters and radiation detectors.

(4) Overall System Design and Operation

The main methods for detection of leakage from the RCPB and the core cooling systems in the UK ABWR are summarised as follows.

(a) Detection of Leakage within the PCV

The leakage inside the PCV is largely divided into unidentified leakage from unknown sources and identified leakage from known sources.

(i) Unidentified Leakage

The primary detection methods for small unidentified leaks within the D/W include (1) D/W High Chemical Impurities Waste System (HCW) sump pump activity and sump level increases, (2) DWC condensate flow rate increases, and (3) D/W airborne particulate activity increases. These variables are continuously indicated and/or recorded in the MCR.

The sensitivity of the detection equipment of these primary detection methods for unidentified leakage within the D/W is sufficient to detect a leakage rate of approximately 3.8L/min within one hour (sensitivity is 3.8L/min in one hour) for (1) and (2). This value is set based on the recommendations from US NRC Regulatory Guide. It is low enough to allow time for corrective action before the process barrier could be significantly compromised allowing for the possibility that most of the unidentified leakage rate might be emitted from a single crack in the nuclear reactor coolant system barrier. There is no particular leakage rate detection requirement for (3).

If the unidentified leakage increases beyond this set point, the detection instrumentation channel will initiate an alarm in the MCR to alert the operator and shut down the reactor when necessary.

As secondary detection methods, pressure and temperature of the D/W atmosphere are used to detect gross unidentified leakage. High D/W pressure will alarm and trigger the isolation logic, which will result in closure of the PCV isolation valves. High D/W temperature is recorded and alarmed only.

(ii) Identified Leakage

The D/W Low Chemical Impurities Waste System (LCW) collects drainage from equipment. Hence, the detection of small identified leakage within the D/W is accomplished by monitoring of LCW sump pump activity and increases in the sump level. LCW sump pump activity and sump level increases will be caused primarily from leaks from large process valves through the valve gland leak-off lines.

The LCW sump level monitoring instruments will activate an alarm in the MCR when the identified leak rate reaches approx. 95L/min within one hour (sensitivity is 95L/min in one hour). The limit of 95L/min for identified leakage within the PCV is a sufficiently low rate set not to affect the correct operation of the plant after considering the available coolant makeup capacity by the CRD purge system and the drainage capacity of the sumps.

The determination of the source of other identified leakage within the D/W is accomplished by (1) monitoring the RPV head seal drain line pressure, (2) monitoring temperature in the valves gland seal leak-off line to the equipment collecting sumps, and (3) monitoring temperature in the SRV discharge lines to the S/P to detect leakage through each of the SRVs. All of these monitors continuously indicate and/or record in the MCR and will initiate an alarm in the MCR on detection of leakage from monitored components.

(iii) Large Leakage

Large leakage inside the D/W (e.g. process line break or LOCA) is detected by high D/W pressure, low reactor water level, or high MS line flow (for break downstream of the flow elements). The instrumentation channels for these variables will initiate an alarm and trigger the isolation logic to close the appropriate isolation valves when the monitored variables exceed predetermined limits, which are different from the previous leakage rates.

In principle, automatic isolation is only triggered in the case of large leakage.

(b) Detection of Leakage outside the PCV

(i) Small Leakage

The areas outside the PCV (D/W) that are monitored for primary coolant leakage are (1) the equipment areas in the R/B, (2) the MS piping vicinity within the R/B, (3) the MS piping vicinity within the Control Building (C/B) and (4) the MS piping vicinity within the T/B. The process piping, for each system to be monitored for leakage, is located in compartments or rooms separated from other systems, so that leakage may be detected by area temperature monitors.

The areas are monitored by thermocouples that sense high ambient temperature in each area. The temperature elements are located or shielded so that they are sensitive to air temperature only and not radiated heat from hot piping or equipment. Increases in ambient temperature will indicate leakage of reactor coolant into the area. These monitors have sensitivities suitable for detection of reactor coolant leakage into the monitored areas. The temperature set points will be a function of the room size and the type of ventilation provided. These monitors provide alarm and indication in the MCR and will trigger the isolation logic to close the appropriate isolation valves (e.g. the MS piping temperature monitors will close the MSIVs, the MS drain isolation valves, and the CUW isolation valves).

In addition to temperature room monitoring, differential ventilation temperature monitoring is provided in equipment areas of the R/B and the MS tunnel room within the R/B to monitor for small leaks. The leakage is monitored and alarmed in the MCR.

(ii) Large Leakage

Large leaks external to the PCV (e.g. process line break outside of the D/W) are detected by low reactor water level, high process line flow, high ambient temperatures in the MS piping vicinity and process equipment areas (e.g. RHR room), high differential flow (CUW only), low MS piping pressures or high RCIC turbine discharge rupture disk pressure. These monitors provide alarm and indication in the control room and will trigger the isolation logic to cause closure of appropriate system isolation valves.

The R/B sump level monitoring instruments will activate an alarm in the MCR when the leakage rate reaches approx. 18.9L/min. The limit of 18.9L/min is a sufficiently low rate set not to affect the correct operation of the plant after considering the available drainage capacity of the sumps.

In principle, automatic isolation is only triggered in the case of large leakage.

(iii) Intersystem Leakage

Intersystem leakage detection is accomplished by monitoring radiation of the Reactor Building Cooling Water System (RCW) coolant return lines from the RIPs, the RHR, the CUW, and the Fuel Pool Cooling and Clean-up System (FPC) heat exchangers. This monitoring is provided by the Process Radiation Monitoring System (PRM).

Design Bases

This section describes the design bases for the LDS. The LDS has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

Normal and Fault Conditions

(1) The LDS is a principal means to deliver monitoring of leakage within the reactor coolant system. [LDS SFC 4-3.1]
(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

Fault Conditions

(2) The LDS is a principal means to deliver detection and alarm of leakage from the reactor coolant system as well as initiation of the signals to isolate the corresponding systems in the event a leakage is detected. [LDS SFC 4-7.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

System Design Description

This section describes the design of the LDS to support and justify the delivery of [LDS SFC 4.3-1] and [LDS SFC 4.7-1]. Additional design description can be found in Ref 12.3-2, Ref 12.3-13, Ref 12.3-14 and Ref 12.3-15.

(1) Leak Detection and Alarm Function

The LDS monitors the following parameters for detection and alarm of leakage from the reactor coolant systems before the leakage leads to the need for PCV isolation:

- PCV temperature,
- D/W airborne particulate activity,
- DWC cooling unit condensed water flow rate,
- D/W waste sumps inflow rate,
- R/B liquid waste sump water discharge flow rate,
- components/piping room ventilation inlet/outlet ΔT, and
- others.

(a) Measurement of the PCV Temperature

The temperature of the atmosphere inside the PCV is controlled and adjusted by the DWC Cooling Unit. In the event that the temperature exceeds the normal operating range, reactor water or steam is considered to be leaking and an alarm is provided in the MCR. The following parameters are monitored for temperature increases in the PCV and an alarm is provided to the MCR in the case of “high” temperature detected.

(i) Atmosphere temperature at various elevations in the PCV (detected by the LDS).

(ii) Air inlet/outlet temperature of the DWC Cooling Unit (detected by the Atmospheric Control System (AC)).

(iii) Temperature difference between the cooling water inlet/outlet of the DWC Cooling Unit (detected by the LDS).

(iv) Temperature difference between the cooling water inlet/outlet of the DWC HVAC Normal Cooling Water System (HNCW) Cooling Unit (detected by the LDS).

(b) Measurement of the D/W Airborne Particulate Activity

The PRM continuously monitors the particulate activity of the atmosphere in the D/W to detect the increase of radioactive substance in the airborne which could come from leakage of reactor water or steam. The PRM monitors and records the data in the MCR initiating an alarm in case of high radiation level.

(c) Measurement of the DWC Cooling Unit Condensate Flow Rate

Steam leaked into the D/W is condensed by the DWC Cooling Unit and collected by the HCW sump in the D/W.

The LDS is provided with a leakage flow meter to measure the flow rate of DWC cooling unit condensate to the HCW sump. The flow detection equipment has a sensitivity of 3.8L/min within one hour for the detection of unidentified leakage as condensate from the DWC cooling unit.

The measurements are recorded and an alarm is provided to the MCR in the case of high flow rate above 3.8L/min within one hour was detected.

(d) Measurement of the Flow Rate into the Drywell Liquid Waste Sump

(i) D/W LCW Sump

Leakage whose origin can be clearly identified such as drains from the RPV flange seal, valve gland seal and other component drains are collected in the D/W LCW sump.

The Radioactive Drain Transfer System (RD) is provided with sump water level meters that measure the water level increase rate in order to monitor the drains flow rate into the D/W LCW sump and detected any potential leakage.

In addition, the drain flow rate into the LCW sump is monitored by measuring the LCW sump pump operation time and intervals.

If the drain flow rate into the sump increases above 95L/min within one hour an alarm is initiated in the MCR to alert about abnormal high flow rates that could be due to leakage.

(ii) D/W HCW Sump

The HCW sump collects high conductivity drains from the RCW, condensate from the DWC Cooling Unit and other leakage from the sources of which cannot be identified (floor drains, etc.).

The RD is provided with sump water level meters that measure the water level increase rate in order to monitor the drains flow rate into the D/W HCW sump and detect any potential leakage.

In addition, the drain flow rate into the HCW sump is monitored by measuring the HCW sump pump operation time and intervals.

The RD HCW sump water level meter has a sensitivity of 3.8L/min within one hour for the detection of unidentified leakage that could flow into the HCW sump.

If the drain flow rate into the sump increases above 3.8L/min within one hour an alarm is initiated in the MCR to alert about abnormal high flow rates that could be due to leakage.

(e) Measurement of R/B Liquid Waste Sumps Drain Flow Rate

The R/B liquid waste sumps collect leakage from the reactor coolant systems inside the R/B.

The RD is provided with instrumentation to measure the R/B sumps pump operation time and intervals in order to monitor the drains flow rate into the sumps and detect any potential leakage.

If the drain flow rate into the sumps increases above 18.9L/min an alarm is initiated in the MCR to alert about abnormal high flow rates that could be due to leakage.

(f) Components/piping Room Ventilation Inlet/outlet ΔT

The LDS is provided with temperature detectors to monitor the differential temperature between the ventilation inlet and outlet ducts in the rooms indicated below in order to detect any leakage in the atmosphere.

The detected data are recorded in the MCR and an alarm is initiated in case of high temperature difference.

(i) MS Tunnel Room within the R/B,
(ii) RHR Pump and Heat Exchanger Room,
(iii) RCIC Pump and Piping Rooms, and
(iv) CUW Regenerative Heat Exchanger and Valve Room.

(g) Others

(i) Measurement of the Pressure between RPV Flange Seals

The NB is provided with leakage detection piping connected between the double seal of the RPV flange. In the event that a pressure increase was detected in the piping and alarm is initiated in the MCR to alert about a potential leakage from the first seal.

(ii) Measurement of Temperature in the RPV Vent Line

The NB is provided with instrumentation to monitor the temperature in the RPV head vent lines in order to detect leakage from the RPV vent valves during reactor operation. The measured data are recorded and an alarm is initiated to the MCR to inform that the set point of temperature is exceeded.

(iii) Measurement of the Outlet Temperature in the SRVs

The NB is provided with instrumentation to monitor the temperature in the SRVs steam discharge line in order to detect potential leakage from the SRVs during reactor operation. The temperature detectors are appropriately separated from their respective valve bodies in order to prevent heat transfer effects from them. The measured values are recorded and an alarm is initiated to the MCR if the set point of temperature is exceeded.

(2) Leak Detection and Isolation Function

There are many monitoring parameters that provide indication of loss of reactor coolant inventory inside and outside of PCV for the delivery of HLSF 4-7 (refer to the 'Primary Containment Isolation System Design Philosophy' [Ref 12.3-30] for details).

The LDS monitors the following parameters for detection and alarm of leakage from the RCS as well as initiation of the signals to close the associated PCV isolation valves:

- Reactor Water Level.
- Component/piping Room Temperature:
  - MS Piping Vicinity (R/B, C/B and T/B),
  - RHR Pump and Heat Exchanger (Hx) Room,
  - RCIC Pump and Piping Rooms, and
  - CUW Regenerative Hx and Valve Room.
- Flow Rate and Pressure in Process Pipes:
  - MS Piping,
  - RCIC Piping, and
  - CUW Piping.
- D/W Pressure.

The LDS measures diverse parameters for defence-in-depth but it only uses one parameter as the principal signal to quickly close the PCV isolation valves and satisfy the dose targets as claimed in the safety analysis. Details are described in the 'Primary Containment Isolation System Design Philosophy' [Ref 12.3-30].

<u>Principal Parameters Used as Signals for Isolation</u>

(a) Measurement of the Reactor Water Level

A decrease in reactor water level caused by reactor water leakage is monitored and detected by the Reactor Pressure Vessel Instrument System (RVI).

The RVI will initiate the alarms and isolation signals at reactor water levels L3, L2, L1.5 and L1 to close the associated isolation valves. The reactor water level is the principal signal for isolation of the CUW and RHR when the leakage comes from their own piping as well as all the RCPB and non-RCPB piping penetrating the PCV when there is leakage from other RCPB pipe inside the PCV.

(b) Flow Rate and Pressure in Process Pipes

(i) Measurement of MS Piping Flow Rate and Pressure

The RVI is provided with four differential pressure transmitters which are installed on the MS Flow Restrictors to detect MS piping ruptures and initiate the MSIVs and MS drain line isolation valves closing signal in the event of high flow rate. The MS high flow rate is the principal signal for isolation of the MS when the leakage comes from its own piping. The MS is provided with pressure detectors on each MS pipe on the turbine side to detect MS pressure drops and initiate isolation of the MSIVs and MS drain line isolation valves.

(ii) Measurement of RCIC Piping Flow Rate and Pressure

The RCIC is provided with differential pressure meters and pressure detectors on the RCIC steam piping elbow. In the case that the detectors indicate excessive differential flow or low pressure in the steam piping, an alarm and the isolation signal to close the RCIC isolation valves is initiated. The RCIC high flow rate is the principal signal for isolation of the RCIC when the leakage comes from its own piping break outside the PCV.

Additional Diverse Signals not Claimed in the Safety Analysis to meet Dose Targets

(a) Component/piping Room Temperature

(i) Measurement of the Temperature in the MS Piping Vicinity

The LDS is provided with temperature detectors to record the data and detect any leakage in the atmosphere around the MS piping inside the MS tunnel room within the R/B, the C/B and the T/B up to the Main Stop Valve (MSV) (the later outside the MS tunnel room).

The detected data are recorded in the MCR and an alarm is initiated in case of high atmosphere temperature. Moreover, the signal to close the MSIVs and the MS drain line isolation valves is initiated in the case that further increase of atmosphere temperature is detected. The MS piping vicinity temperature is just an additional diverse signal for defence-in-depth, but is not claimed in the safety analysis as principal signal for isolation of the MS piping.

(ii) Measurement of RHR Pump and Heat Exchangers Room Temperature

The LDS is provided with temperature detectors in the areas where RHR Pumps and Heat Exchangers are located in order to detect the atmosphere temperature and to record it for the detection of any leakage. An alarm is initiated in the case of high atmosphere temperature. Moreover, the signal to close the RHR isolation valves is initiated in the case that further increase of atmosphere temperature is detected. The RHR equipment room temperature is just an additional diverse signal for defence-in-depth, but is not claimed in the safety analysis as principal signal for isolation of the RHR piping.

(iii) Measurement of RCIC Pump and Piping Rooms Temperature

The LDS is provided with temperature detectors in the areas where RCIC Pump and pipes are located in order to detect the atmosphere temperature and to record it for the detection of any leakage. An alarm is initiated in the case of high atmosphere temperature. Moreover, the signal to close the RCIC isolation valves is initiated in the case that further increase of atmosphere temperature is detected. The RCIC equipment room temperature is just an additional diverse signal for defence-in-depth, but is not claimed in the safety analysis as principal signal for isolation of the RCIC piping.

(iv) Measurement of CUW Components Rooms Temperature

The LDS is provided with temperature detectors in the areas where CUW regenerative heat exchanger and valves are located in order to detect the atmosphere temperature and to record it for the detection of any leakage. An alarm is initiated in the case of high atmosphere temperature. Moreover, the signal to close the CUW isolation valves is initiated in the case that further increase of atmosphere temperature is detected. The CUW equipment room temperature is just an additional diverse signal for defence-in-depth, but is not claimed in the safety analysis as principal signal for isolation of the CUW piping.

(b) Flow Rate and Pressure in Process Pipes

Measurement of CUW Piping Flow Rate

Leakage is detected by the flow rate difference which is detected by comparing the inflow and outflow through the CUW. The CUW is provided with instruments for measuring the flow rate difference that are capable of the fluid density correction for fluid temperature changes, which is required to calculate correct volume flow.

A set point is determined within a range in order not to initiate false signals due to instrumentation errors. An alarm and the isolation signal to close CUW isolation valves are initiated in the event that the flow difference between the inflow and the outflow exceeds the set point. The CUW differential flow rate is just an additional diverse signal for defence-in-depth, but is not claimed in the safety analysis as principal signal for isolation of the CUW piping.

(c) D/W Pressure
The pressure in the PCV is controlled in plant normal operation conditions by the AC; however, the containment is subjected to continuous pressure variations due to factors such as air pressure changes, leakage, etc. The PCV pressure changes exceeding the range in plant normal operation condition are considered as a leakage from a system.
The AC is provided with instrumentation to monitor and detect these leakages and initiate an alarm in the MCR in conjunction with the RVI. If the D/W pressure exceeds the limits, the containment isolation signal is also triggered. The D/W pressure is just an additional diverse signal for defence-in-depth, but is not claimed in the safety analysis as principal signal for isolation of any pipe.

(3) Main Support Systems
(a) Instrumentation and Control Systems

Instrumentation and controls are provided to measure and control the operational conditions of the LDS components necessary for the delivery of the safety functions claimed. The relevant instrumentation and control provisions are described as follows. For further details refer to Chapter 14: Control and Instrumentation, Section 14.6 Control and Instrumentation Systems.

(i) Leak Detection and Alarm Function
- The PCV temperature is monitored by the LDS and AC instrumentation.
- The D/W airborne particulate activity is monitored by the PRM.
- The DWC cooling unit condensate flow rate and the components/piping room ventilation inlet/outlet differential temperature are monitored by the LDS.
- The D/W sumps and R/B sumps flow rate is monitored by the RD.
- The pressure between the RPV flange seals, the temperature in the RPV vent line and the temperature in the SRV discharge line are monitored by the NB.
- The temperature in the valve gland leak-off line is monitored by the VGL.
- The leak monitoring and alarms are processed by the Reactor Auxiliary Control System.
- The LDS portion for this function is continuously in operation and the indicators, recorders and alarms are provided in the MCR HMI.
- Abnormal leakage from any of the systems included in the RCPB is detected by at least two different methods for each location.

(ii) Leak Detection and Isolation Function
- The reactor water level and MS piping flow rate are monitored by the instrumentation of the RVI. The MS piping pressure is monitored by the MS instrumentation on the T/B side.
- The temperature in equipment rooms is monitored by LDS instrumentation.
- The RCIC piping flow rate/pressure and RCIC turbine discharge rupture disk interspace pressure are monitored by the RCIC instrumentation.
- The CUW piping differential flow is monitored by the CUW instrumentation.
- The D/W pressure is monitored by the AC and the RVI instrumentation.
- The alarm and isolation signals are processed by the Safety System Logic and Control System (SSLC) logic.

- The Primary Containment Isolation System (PCIS) receives the signals from the LDS, RVI, MS, RCIC, CUW and AC and initiates primary containment isolation by sending the closure signals to the valves of the corresponding system through the SSLC logic.
- The LDS is continuously in operation and the indicators, recorders and alarms are provided in the MCR HMI.
- Abnormal leakage from any of the systems included in the RCPB is detected by at least two different methods for each location.
- Isolation of the containment due to detection of leakage from the reactor coolant systems is done by at least two different methods.
- Instruments and signal systems for isolation are integrated into the safety protection systems and consist of 4 divisions with 2 out of 4 logic configuration.

(b) Power Supply System

(i) Leak Detection and Alarm Function
Power to the instrumentation for the detection of leakage and alarm is supplied by the switchboard supplied from the Safety Class 1 Uninterruptible Power Supply (UPS).
Even though it is not required, in the event of Loss of Offsite Power (LOOP) power supply is backed up by the Emergency Diesel Generators (EDGs) as defence-in-depth measure.

(ii) Leak Detection and Isolation Function
Power to the instrumentation for the detection of leakage and transmission of isolation signals is supplied by functionally independent and separated divisions of the Safety Class 1 UPS. In the event of LOOP the power supply is backed up by the EDGs.

Assumptions, Limits and Conditions for Operation

In order to ensure that the LDS is operated within the safety limits and that the design requirements from the safety case are met during the operating regime, appropriate LCOs and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the ‘Basis of Safety Cases on Reactor Coolant Pressure Boundary and Core Cooling Systems Leakage Detection System’ (Ref 12.3-2).

- Unidentified leakage shall be within 3.8L/min in one hour during start-up, power operation and hot shutdown for the delivery of the SFCs claimed when required.
- Total leakage shall be within 98.8L/min in one hour during start-up, power operation and hot shutdown for the delivery of the SFCs claimed when required.
- D/W HCW and LCW sumps monitoring subsystem shall be operable during start-up, power operation and hot shutdown for the delivery of the SFCs claimed when required.
- The low pressure portion shall not be overpressurised due to leakage from the reactor coolant system isolation valves during start-up, power operation and hot shutdown for the delivery of the SFCs claimed when required.

12.3.4 Reactor Vessel

12.3.4.1 Reactor Pressure Vessel

Design Bases

(1) The RPV is designed to assure the structural integrity in order to maintain the reactor coolant volume to ensure appropriate core cooling or residual heat removal capabilities during all operating conditions.

(2) The RPV is designed to assure structural integrity of attachment supports for the reactor internals in order to deliver the safety functions required for the reactor internals during all operating conditions.

The components of the RPV main pressure boundary are classified as VHI. VHI is assigned where it is impracticable to protect against gross failure, the consequences of which are intolerable. The severity of consequences of failure varies by region and not all regions of the RPV are classified as VHI; regions where consequences of failure are tolerable and protection exists are assigned Standard Class 1.

The RPV is a cylindrical welded steel structure that contains the core, reactor internals and light water coolant. The RPV consists of a removable and flanged top head, cylindrical shell, bottom head and various nozzles.

The RPV is a cylindrical shell with rounded ends, approximately 21m tall, with an outer diameter of approximately 7.5m. The cylindrical shell is constructed from four low alloy steel ring forgings, joined by circumferential welds only. The nominal thickness of the shell rings is approximately 175mm. The top ring features 8 total penetrations for the MS piping and first set of water level detectors. The second ring features 17 penetrations for another set of water level detectors, LPFL, HPCF, Shutdown outlet nozzles and FDW lines. The third ring only features 4 penetrations for the last set of water level instrumentation and the Bottom ring has no penetrations. All of the major penetrations are located above the top of the core, reducing the probability of core damage. The MS line nozzles feature a venturi-type flow restrictor and the FDW, LPFL and HPCF nozzles utilise thermal sleeve welded to the nozzles, protecting the vessel's inner blend radius against the effects of high fatigue due to thermal cycling.

The RPV is supported by a conical support skirt attached to the lower vessel cylindrical shell. The support skirt attachment is machined from the forging ring, removing the requirement for a support skirt attachment weld from the cylindrical shell. This design provides enhanced lateral support to the RPV body in the event of an earthquake and improves maintenance access to the RIPs. In addition, to support lateral force due to seismic load, the upper part of the RPV shell is supported horizontally by the RPV stabilizers on the reactor shield wall. The RPV support skirt is secured to the RPV pedestal with anchor bolts.

The top head is hemispherical, whereas the bottom head is torispherical. The top head weighs approximately 100 tonnes and is designed to be removable during operational life for refuelling. To facilitate this, four welded lugs allow lifting equipment to be attached. The top head flange is secured to the RPV shell flange using 80 main closure bolts, each approximately 1.75m long. The seat surface of the RPV flange is a double-seal structure of metallic O-rings for leak prevention. Leak detection instrumentation monitors the interspace between the two seals to detect leakage from the inner seal.

The bottom head consists of a spherical dome made from a single forging, and a conical transition section to a toroidal knuckle between the bottom head dome and vessel shell, also made from a single forging. The bottom head dome features over 260 penetrations, 205 of which are for the Control Rod Drives (CRDs) and 62 for In-core monitoring equipment. The toroidal knuckle forging includes the penetrations for the 10 RIPs. Surveillance test coupons are to be installed in the RPV for periodic surveillance and analysis of changes in mechanical properties of RPV material due to neutron irradiation during the operating period.

Safety Evaluation

The standards by which structural integrity is assured reflect the functional reliability requirements of the RPV, commensurate with its VHI classification. Chapter 8: Structural Integrity, Section 8.5 Components Safety Reports identifies component safety reports and describes the approach for each classification.

The Reactor Pressure Vessel safety case in terms of the structural integrity is discussed in the 'Topic Report on RPV Structural Integrity' (Ref 12.3-26).

Assumptions, Limits and Conditions for Operation

In order to ensure that the RPV, as part of the RCS is operated within the safety limits and that the design requirements from the safety case are met during the operating regime, appropriate LCOs and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the 'Topic Report on RPV Structural Integrity' (Ref 12.3-26).

- The RCS pressure and temperature shall be in accordance with the limits specified in Ref 12.3-26 at all times.
- The reactor pressure shall be below the limit specified in Ref 12.3-26 during start-up and power operation for the delivery of the SFCs claimed when required.

12.3.4.2 Reactor Internals

Design Bases

(1) The core support structures within the reactor internals are designed to assure the structural integrity in order to maintain vertical upward and horizontal support and positioning for the fuel assemblies to ensure flow paths for appropriate core cooling or residual heat removal and to maintain the structure to ensure the rapid insertion of CRs to achieve sub-criticality during all operating conditions. Loss of the coolable core geometry could lead to consequences above the BSL. From this perspective, the core support structures within the reactor internals deliver a Category A safety function and the components necessary to deliver this function are classified as Class 1 safety components according to the safety categorisation and classification of UK ABWR, defined in Chapter 5: General Design Aspects, Section 5.6.

(2) CRGTs within the reactor internals are designed to assure the structural integrity in order to provide guidance for the rapid insertion of CRs to achieve sub-criticality during all operating conditions. Loss of the function of rapid CR insertion would potentially lead to consequences above the BSL. From this perspective, the CRGTs within the reactor internals deliver a Category A safety function and the components necessary to deliver this function are classified as Class 1 safety components according to the safety categorisation and classification of UK ABWR.

(3) Feedwater spargers, LPFL spargers, HPCF couplings and HPCF spargers within the reactor internals are designed to assure the structural integrity in order to maintain appropriate core cooling or residual heat removal capabilities by the ECCS during design basis faults. Loss of the core cooling or the residual heat removal functions would lead to consequences above the BSL. From this perspective, the feedwater spargers, LPFL spargers, HPCF couplings and HPCF spargers within the reactor internals deliver a Category A safety function and the components necessary to deliver this function are classified as Class 1 safety components according to the safety categorisation and classification of UK ABWR.

(4) HPCF couplings and HPCF spargers within the reactor internals are designed to assure the structural integrity in order to inject sodium pentaborate solution into the core to achieve and maintain sub-criticality by the SLC in case of CRs insertion failure during all operating conditions. Loss of the alternative reactivity control function would potentially lead to consequences above the BSL. From this perspective, the HPCF couplings and HPCF spargers within the reactor internals deliver a Category A safety function and the components necessary to deliver this function are classified as Class 2 safety components according to the safety categorisation and classification of UK ABWR. However, the HPCF couplings and HPCF spargers are classified as Class 1 safety components based on above (3).

Facility Design

The facility design of reactor internals is described in section 12.3.1.2. The Reactor Internals safety case in terms of the structural integrity is discussed in ‘Topic Report on Reactor Internals’ (Ref 12.3-25).

Assumptions, Limits and Conditions for Operation

There are no relevant LCOs for the reactor internals because the conditions for operation of the RIN are bounded by the RCS LCOs defined in section 12.3.4.1.

12.3.5 Components and Subsystems

12.3.5.1 Reactor Recirculation System (RRS)

Summary Description

This section is a general introduction to the RRS where the system roles, system functions, system configuration and modes of operation are briefly described. The RRS safety case is justified in the 'Basis of Safety Cases on Reactor Recirculation System' (Ref 12.3-3). The RRS design is described in detail in the system design specifications (Ref 12.3-22) and the P&IDs (Ref 12.3-23).

(1) System Roles

The RRS is designed for the following purposes from the power generation perspective:

(a) To force the reactor coolant to circulate through the reactor core in order to transfer the heat generated by the nuclear fission reaction in the core to the coolant.

(b) To adjust the reactor core flow rate to control the reactor power.

(2) Functions Delivered

The RRS is designed to perform the following functions for power generation:

(a) The RRS is provided with 10 RIPs installed at the bottom of the RPV to force recirculation of the reactor coolant for heat transfer.

(b) The RRS adjusts the reactor core flow rate to control the reactor power by changing the rotation speed of the RIPs. The rotation speed of the RIPs is controlled by the Recirculation Flow Control System (RFC). Hence, the RRS allows reactor power control without changing the position of the CRs.

(3) Basic Configuration

In addition to the RIP main units, the RRS is configured with the following auxiliary subsystems, shown in Figure 12.3-11.

(a) Recirculation Motor Cooling System (RMC)

This system forces cooling water to circulate between the RIP motor and the RIP Motor Heat Exchanger to cool down the motor. The circulation of the RIP motor cooling water is implemented by the auxiliary impeller mounted on the RIP rotor. The RIP motor cooling water transfers the heat removed from the RIPs to the RCW through the RIP Motor Heat Exchanger.

(b) Recirculation Motor Purge System (RMP)

This system supplies purge water to minimise radiation doses during RIP maintenance by preventing crud in the reactor coolant from flowing into the RIP casing side during operation. The purge water is supplied from the CRD and flows into the reactor through the gap between the shaft and the stretch tube.

(c) Recirculation Motor Inflatable Shaft Seal System (RMISS)

This system pressurises the inflatable seal placed in the casing against the stationary pump shaft to prevent the reactor coolant from leaking into the casing side when the RIP motor is removed for maintenance during plant outage. The seal water is supplied from a portable tank and pump.

Furthermore, control of reactor power through the flow control region by adjustment of the rotation speed of the RIPs is provided by the RFC, which provides properly conditioned control and logic signals, which regulate the reactor coolant recirculation flow rate produced by the RRS under various steady-state, transient, and emergency modes of Nuclear Steam Supply System (NSSS) operation. The following three subsystems are designated as part of the RFC. For further details refer to Chapter 14: Control and Instrumentation, Section 14.6.5 Plant Control System.

(d) Adjustable Speed Drives (ASD)

The ASDs are used to provide electrical power and speed control to the pump motors in the RIPs. Each ASD receives electrical power at a constant AC voltage and frequency. The ASD converts this to a variable frequency and voltage in accordance with the speed demand requested by the RFC controller. The variable frequency and voltage is supplied to vary the rotation speed of the RIPs to adjust the pumps flow rate (reactor coolant flow rate).

(e) Recirculation Pump Trip System (RPT)

In the event of either (a) turbine trip or generator load rejection when reactor power is above a predetermined level, (b) reactor pressure exceeds the high RPV pressure trip set point, or (c) reactor water level drops below a certain set point, the RPT logic will automatically trip off a group of four RIPs.

(f) Core Flow Measurement Systems (CFM)

These systems consist of differential pressure measurements taken across the core plate and the RIP which are processed to provide flow measurements.

(4) Modes of Operation

(a) The 10 RIPs of the RRS continuously operate during start-up and power operation (including temporary hot standby states in the start-up mode as defined in Chapter 5: General Design Aspects, Section 5.4.3 Operating Modes and Plant Operating States) to provide reactor coolant recirculation.

(b) During certain transients, various RIP operating modes will be required, such as: (1) RIPs runback following loss of one reactor Feedwater Pump, (2) trip of selected RIPs from current reactor protection conditions; or runback-to-minimum speed and subsequent trip. These control actions are all produced by the RFC.

(c) The RMC and RMP continuously operate, with the exception of during maintenance and inspection.

(d) The RMISS operates only when the RIP is stopped and the motor is removed for maintenance during plant outage.

Design Bases

This section describes the design bases for the RRS. The RRS has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

<u>Normal Conditions</u>

(1) The RRS in conjunction with the RFC provides reactor coolant forced recirculation for power generation during normal operation conditions, whose failure could lead to a total loss of reactor coolant flow and consequently demand a Category A safety function to mitigate it. [RRS SFC 1-6.1]

(This prevention function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements)

(2) The RRS in conjunction with the RFC provides reactor coolant forced recirculation for power generation during normal operation conditions, whose failure could lead to a partial loss of reactor coolant flow and consequently demand a Category A safety function to mitigate it depending on the case. [RRS SFC 1-6.2]

(This prevention function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements)

<u>Normal and Fault Conditions</u>

(3) The RRS portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [RRS SFC 4-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified as explained in section 12.3.3.1)

<u>Fault Conditions</u>

(4) The RIPs of the RRS are tripped by the RPT by a signal from the Hardwired Backup System (HWBS) as part of the actions to perform alternative shutdown of the reactor in the event of Anticipated Transient Without Scram (ATWS). [RRS SFC 1-5.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements (RRS components are designed to meet Safety Class 3 requirements). Refer to Chapter 14: Control and Instrumentation, Section 14.6.3 Hardwired Backup System for further details)

(5) The RIPs of the RRS are tripped by the RPT by a signal from the Plant Control System (PCntlS) as part of the actions used to deliver mitigation of power increases. [RRS SFC 1-8.1]
(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements. Refer to Chapter 14: Control and Instrumentation, Section 14.6.5 Plant Control System for further details)

(6) The RRS components penetrating the primary containment form a barrier to confine radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults. [RRS SFC 4-7.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in Chapter 13: Engineered Safety Features, Section 13.3.3.2 Primary Containment Isolation System)

System Design Description

This section describes the design of the RRS to support and justify the delivery of [RRS SFC 1-6.1] and [RRS SFC 1-6.2]. Additional design description can be found in Ref 12.3-3, Ref 12.3-22 and Ref 12.3-23.

(1) Overall System Design and Operation

The RRS provides reactor coolant recirculation through the RIPs to transfer the heat generated by the nuclear fission reaction in the core to the coolant. Failure of the RRS SSCs would lead to partial loss of recirculation flow or a total loss depending on the number of components that failed.

Partial loss of reactor recirculation flow produces a mild transient of flow and power level. From the radioactive dose perspective, unmitigated partial loss of reactor recirculation flow has very small consequences, resulting in a radioactive dose below the Basic Safety Objective (BSO) in the safety analysis presented in Chapter 24: Design Basis Analysis.

Total loss of recirculation flow produces a larger loss of reactor flow and higher increase in power level and may cause some fuel cladding heat up due to short-term transition boiling. From the radioactive dose perspective, unmitigated total loss of reactor recirculation flow also has very small consequences, resulting in a radioactive dose below the BSO in the safety analysis.

In order to prevent failure, the RRS is provided with instrumentation that controls the process parameters and informs the operators in the MCR about the system state before any abnormality leads to a fault. Should a failure occur, the instrumentation and control provisions would detect it promptly so that the corresponding mitigation measures can be put in place.

In the event of a partial loss of recirculation flow, the RFC would detect the loss of the RIPs and initiate the Select Control Rod Run-In (SCRRI) action in order to mitigate its effects (scram action and isolation are not necessary). Plant operation at low power can be continued in this event. However, if considered appropriate, the reactor can be manually shutdown by scram followed by RCIC actuation as auxiliary feedwater mode at reactor water Level-2 (reactor water levels are defined in the 'Topic Report on Reactor Pressure Vessel Instrument System' [Ref 12.3-31]) to bring the reactor to hot shutdown. Depressurisation by the SRVs and RHR operation follow to provide long-term heat removal to bring the reactor to cold shutdown.

In the event of a total loss of recirculation flow, the Reactor Protection System (RPS) would initiate the mitigation actions. The RPS would initiate scram action after detection of a rapid core flow coastdown or high thermal power output. After scram, the RCIC auxiliary feedwater mode is initiated to restore and maintain the reactor coolant inventory when water level drops to Level-2. Finally, depressurisation by the SRVs and RHR operation is implemented to provide long-term heat removal to bring the reactor from hot shutdown to cold shutdown.

(2) Equipment Design and Operation

(a) RIPs

(i) Purpose

The RIPs have two required functions. One of the functions is to force recirculation of the reactor water to the reactor core, in order to transfer the heat generated in the reactor core to the reactor water effectively. The second function is to change the flow rate of the reactor water by controlling the rotational speed of the RIPs, in order to control the reactor thermal output power when in operation. These two functions are related to the delivery of [RRS SFC 1-6.1] and [RRS SFC 1-6.2].

(ii) Configuration and Operation

The RRS incorporates an arrangement of ten reactor coolant recirculation pump units referred to as RIPs. The RIPs are installed at the bottom of the RPV with their impellers and diffusers internal to the RPV. The RIPs themselves are mounted vertically onto and through the RIP nozzles that are arranged equally-spaced on the bottom head of the RPV. The RIPs are single-stage vertical pumps with induction wet type motor to reduce leakage potential further by eliminating the shaft seal part.

The RIPs provide forced circulation of the reactor coolant through the lower plenum of the reactor and up through the core plate, the reactor core, top guide, the steam separators, and back down the downcomer annulus. The ten RIPs are designed such that even in the event of single failure of one RIP the recirculation flow can be maintained at full reactor power in most circumstances with the remaining nine RIPs.
The RIP motors are the variable speed, four-pole, AC induction wet motor type. The operating speed of the motor depends on the variable-frequency output of the ASDs. The RIP motors are cooled by water from the primary side of the RIP Motor Heat Exchangers. Heat in the secondary side of the heat exchanger is removed by the RCW. There is one heat exchanger per motor.
The motor casing which houses the RIP motor is welded to the bottom of the RPV.

Since there is no reactor coolant recirculation piping, pump run-out phenomena due to the recirculation piping break does not happen. In addition, the absence of any large bore piping below the top of active fuel ensures that the core can remain covered in the event of LOCA, by using the ECCS to replace any limited loss of coolant inventory.

(b) RIP Motor Heat Exchanger

(i) Purpose

The RIP Motor Heat Exchanger is intended to transfer the heat from the RIP motor cooling water to the reactor building cooling water in order to cool the RIP motor. It directly supports the RIP for delivery of its power generation functions. This function is related to the delivery of [RRS SFC 1-6.1] and [RRS SFC 1-6.2].

(ii) Configuration and Operation

The RIP Motor Heat Exchanger is a vertically-oriented, shell-and-tube U-tube heat exchanger with a bottom channel.

The RMC primary coolant from the RIP motor cavity flows outbound from a nozzle near the top of the motor casing, and through piping, which courses across and upward to the RIP Motor Heat Exchanger primary coolant inlet nozzle located near the top of the heat exchanger shell. In moving downward through the shell, this primary coolant sweeps back and forth across the tube bundles guided by horizontal flow baffle/tube-support plates. Flow exits from the heat exchanger shell through a nozzle located just above the tube sheet and crosses, via piping, directly back to the RIP motor casing. Upon entering the RIP motor casing, this primary coolant is drawn into the suction region of the auxiliary impeller, where it is then driven upward through the motor to begin another circuit around this motor heat exchanger - motor flow loop.

(3) Main Support Systems

(a) Instrumentation and Control System

Instrumentation and control are provided to measure and control the operating conditions of the RRS components necessary for the delivery of the reactor coolant recirculation function. The relevant instrumentation and control provisions to detect damage or failure of the RRS are described as follows.

(i) RIP Rotational Speed (RFC)
The rotational speed of the RIPs is controlled by the RFC and displayed in the MCR to control its operation.

(ii) RIP Motor Casing Vibration (RRS)
The vibration of the RIP motor casing is monitored for each RIP in two perpendicular directions for surveillance of the operating status and displayed in the MCR. An alarm is initiated upon high vibrations.

(iii) Recirculation Motor Cooling System (RRS, RFC, RCW, LDS)
The cooling water temperature at the inlet and outlet of the RIP casing is monitored to check whether the cooling is implemented properly and display the result in the MCR. An alarm is initiated upon high outlet temperature and the related RIP is automatically tripped by the RFC.

(iv) Others
In addition the following parameters are monitored by other systems in order to control RRS correct operation.

- Core Flow (RVI),
- Thermal Output (Average Power Range Monitor, APRM), and
- Reactor Water Level (RVI).

The design of the instrumentation and control systems is justified in Chapter 14: Control and Instrumentation, Section 14.6 Control and Instrumentation Systems.

(b) Power Supply System

Power to the RIPs is supplied through the ASDs of the RFC. The ASDs generate variable voltage, variable frequency electrical power for use by the induction motors of the RIPs. There are ten ASDs in total that independently supply power to each RIP. The ASDs receive power input from 4 separate electrical divisions through Motor-Generators (MGs). There are four MGs which supply power to two or three RIPs each such that single failure of one of the buses does not lead to the failure of more than three RIPs. The RIPs supplied by different divisions are arranged around the RPV periphery so that no adjacent RIPs stop simultaneously on failure of a single electrical division.

The design of the power supply system is justified in Chapter 15: Electrical Power Supplies.

Assumptions, Limits and Conditions for Operation

In order to ensure that the RRS is operated within the safety limits and that the design requirements from the safety case are met during the operating regime, appropriate LCOs and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the 'Basis of Safety Cases on Reactor Recirculation System' (Ref 12.3-3).

- The reactor thermal power and core flow shall be within the design power-flow operating region shown in Figure 11.5-11 of Chapter 11: Reactor Core and at least nine RIPs shall be operable during start-up and power operation for the delivery of the SFCs claimed when required.

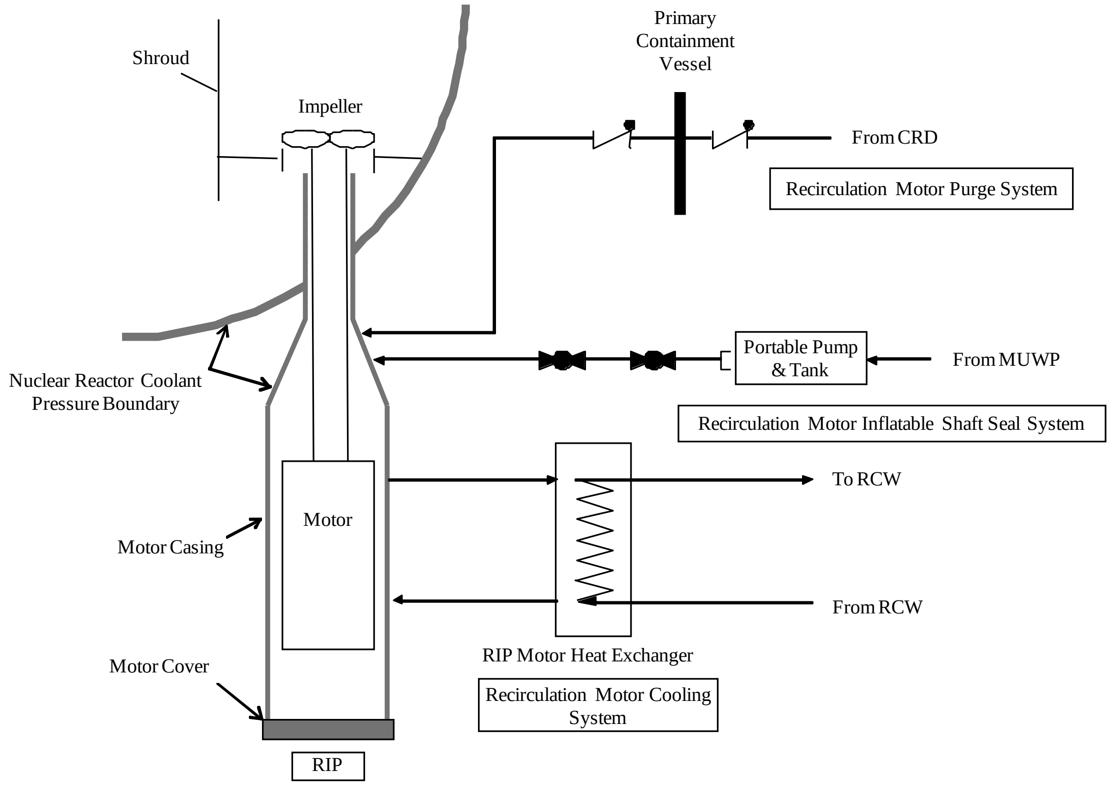

Figure 12.3-11 : Outline of the RRS

12.3.5.2 Nuclear Boiler System (NB)

System Summary Description

This section is a general introduction to the NB where the system roles, system functions, system configuration and modes of operation are briefly described. The NB safety case is justified in the 'Basis of Safety Cases on Nuclear Boiler System' (Ref 12.3-4). The NB is described in detail in the system design specifications (Ref 12.3-8) and the P&IDs (Ref 12.3-9, Ref 12.3-10, Ref 12.3-11 and Ref 12.3-12).

(1) System Roles

The main roles of the NB are to transfer steam generated in the RPV to the turbine system for power generation and also to transfer the condensed steam as feedwater back into the RPV to close the cycle during normal conditions; to provide isolation of the MS to prevent uncontrolled radioactive releases; and to maintain the pressure of the reactor system below specified limits during fault conditions.

(2) Functions Delivered

The NB is designed to perform the following main functions:

(a) The NB transfers steam generated in the RPV to the turbine system to drive the turbines and generate electrical power.

(b) The NB transfers the feedwater from the CFDW into the RPV.

(c) The NB MS lines limit the loss of coolant and the release of radioactive material from the RPV following a MS line rupture outside the PCV.

(d) The NB provides isolation of the MS lines to prevent uncontrolled radioactive releases to the surrounding areas in the event of transients and postulated accidents.

(e) The NB limits the RPV pressure under transient and accident conditions for overpressure protection of the RCPB.

(f) The NB provides reactor core cooling as part of the ECCS as follows:

(i) The SRVs release the steam generated during reactor core cooling by high pressure core cooling systems to maintain their operation.

(ii) The NB depressurises the RPV through the ADS or the RDCF to allow low pressure core cooling.

(iii) The NB provides a piping path so that the RHR can perform low pressure core flooding function through its division A, which is connected to FDW lines A and B to inject water into the RPV.

(iv) The NB also provides a piping path so that the RCIC can perform high pressure core cooling function by supplying water into the RPV through FDW line B. In addition, the steam-driven turbine of the RCIC pump takes steam supply through a branch coming from one MS line.

(g) The NB through the SRVs in conjunction with the RCIC and the RHR Reactor Shutdown Cooling mode provides long-term decay heat removal for reactor shutdown in the event of Main Condenser isolation.

(h) The NB provides piping path so that the CUW can perform clean-up of reactor coolant by returning reactor coolant into the RPV through the FDW lines.

(3) Basic Configuration

The NB is divided into two subsystems, the MS and the FDW. The NB subsystems should be differentiated from the MS and the CFDW of the turbine facility, which are the continuation of the MS and FDW beyond the R/B up to and including the T/B.

(a) MS

The steam generated in the nuclear reactor is transferred from the RPV steam outlet nozzle to the turbine system through four MS lines. These four MS lines join at the upstream header of the turbine MSV in the T/B. The MS consists of the following main components:

- Four MS lines,
- four MS Flow Restrictors (one on each MS line),
- MS drain system,
- 16 SRVs,
- SRV discharge lines,
- eight MSIVs (two on each MS line), and
- instrumentation and controllers.

Figure 12.3-13 shows an outline of the MS.

(b) FDW

The water discharged by the FDW Pumps is transferred to the nuclear reactor through the two feedwater lines. Each feedwater line branches into three pipelines inside the PCV, with a total of six feedwater pipelines connected to the Feedwater Spargers in the RPV to equally supply water into the reactor. The FDW consists of the following major components:

- Two feedwater lines with their associated valves, and
- instrumentation and controllers.

Figure 12.3-12 shows an outline of the FDW.

Design Bases

This section describes the design bases for the NB. The NB has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

Normal Conditions

(1) The MS pipework of the NB outside the RCPB (beyond outboard MSIV) contains reactor coolant and its rupture could lead to a release of radioactive material of dose consequences relatively low, but demanding Category A safety functions to mitigate them. [NB SFC 4-3.1]

(This function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements)

(2) The FDW pipework of the NB outside the RCPB (beyond the outboard check valve) contains reactor coolant and its rupture could lead to a release of radioactive material of dose consequences relatively low, but demanding Category A safety functions to mitigate them. [NB SFC 4-3.2]

(This function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements)

Normal and Fault Conditions

(3) The NB portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [NB SFC 4-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified as explained in section 12.3.3.1)

Fault Conditions

(4) The feedwater flow is controlled by the Hardwired Back-up System logic in order to prevent reactivity insertion in the event of Anticipated Transient Without Scram (ATWS). [NB SFC 1-5.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements. Refer to Chapter 14: Control and Instrumentation, Section 14.6.3 Hardwired Backup System for further details of the functions of the Hardwired Back-up System)

(5) The MS through the SRVs is the principal means to deliver long-term residual heat removal to reach reactor cold shutdown by depressurisation of the RPV in the event of unavailability of the main condenser. [NB SFC 3-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(6) The MS through the SRVs is the principal means to deliver long-term residual heat removal during shutdown in the event of unavailability of the RHR. [NB SFC 3-1.2]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(7) The MS through its flow restrictors is a principal means to limit the loss of reactor coolant and the release of radioactive material from the RPV following a MS line rupture outside the PCV to the extent that the RPV water level does not drop below the top of the active fuel before closure of the MSIVs. [NB SFC 4-7.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(8) The MS is the principal means to close the MS lines to limit the release of reactor coolant and radioactive material to the surroundings in the event of a MS pipe rupture by closing the MSIVs. [NB SFC 4-7.2]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(9) The NB components penetrating the primary containment form a barrier to confine radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults. [NB SFC 4-7.3]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in Chapter 13: Engineered Safety Features, Section 13.3.3.2 Primary Containment Isolation System)

(10) The MS through the SRV discharge line quenchers is a principal means to suppress the dynamic loads generated in the PCV when steam discharged via the SRVs condenses in the suppression pool. [NB SFC 4-7.4]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(11) The MS through the safety valve function of the SRV is the principal means to release the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV. [NB SFC 2-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in section 12.3.3.3 ‘RCPB Overpressure Protection System’)

(12) The MS through the relief valve function of the SRV is an additional means to the release of the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV. [NB SFC 2-1.2]

(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements. This safety function is developed and justified in section 12.3.3.3 ‘RCPB Overpressure Protection System’)

(13) The NB through the ADS is the principal means to depressurise the RPV in order to provide reactor core cooling in low pressure state as part of the ECCS in the event of LOCA infrequent faults inside the PCV. [NB SFC 2-1.3]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System)

(14) The NB through the transient ADS is the principal means to depressurise the RPV in order to provide reactor core cooling in low pressure state as part of the ECCS in the event of LOCA infrequent faults outside the PCV. [NB SFC 2-1.4]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in Chapter 13: Engineered Safety Features, Section 13.4 Emergency Core Cooling System)

(15) The MS through the safety valve function of the SRVs is the principal means to deliver overpressure protection of the RCPB under abnormal transients and accident conditions that could put excessive pressure on the boundary. [NB SFC 4-2.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in section 12.3.3.3 ‘RCPB Overpressure Protection System’)

(16) The MS is a principal means to prevent excessive loss of reactor coolant after SRV actuation for the safety valve function. [NB SFC 4-5.1]

(This function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements. This safety function is developed and justified in section 12.3.3.3 ‘RCPB Overpressure Protection System’)

(17) The MS through the relief valve function of the SRVs is an additional means to deliver overpressure protection of the RCPB faster than the safety valve function under abnormal transients and accident conditions that could put excessive pressure on the boundary. [NB SFC 4-6.1]

(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements. This safety function is developed and justified in section 12.3.3.3 ‘RCPB Overpressure Protection System’)

System Design Description

This section describes the design of the NB to support and justify the delivery of [NB SFC 4-3.1], [NB SFC 4-3.2], [NB SFC 3-1.1], [NB SFC 3-1.2], [NB SFC 4-7.1], [NB SFC 4-7.2] and [NB SFC 4-7.4]. Additional design description can be found in Ref 12.3-4, Ref 12.3-8, Ref 12.3-9, Ref 12.3-10, Ref 12.3-11 and Ref 12.3-12.

(1) Overall System Design and Operation

(a) Containment of Reactor Coolant in MS Pipework beyond the RCPB

The steam generated in the RPV is transferred to the turbine side for power generation through the MS lines. The MS subsystem of the NB consists of four MS pipes, which contain reactor coolant (steam) during start-up, power generation and shutdown. The range of the RCPB goes up to the outboard MSIV as shown on Figure 12.3-13 . The MS piping beyond the outboard MSIV is outside the range of the RCPB, but it still contains coolant that can be lost due to a pipe rupture because of a mechanical failure or the effects of a hazard. Loss of reactor coolant would lead to a decrease in reactor coolant inventory and an increase of radioactive material release into the MS tunnel room.

In order to prevent a fault, the MS is provided with instrumentation that controls the process parameters and informs the operators in the MCR about the plant state before any abnormality leads to a fault such as pipe rupture. Should damage or a pipe rupture occur, the instrumentation and control provisions would detect it promptly so that the corresponding mitigation measures can be put in place.

In order to mitigate an increase of radioactive material release into the MS tunnel room, the LDS would detect any leakage from a damaged MS pipe, initiate an alarm in the MCR and provide the MSIV isolation signals so that the isolation valves are automatically isolated by the PCIS and the loss of coolant is stopped.

In order to mitigate the decrease in reactor coolant inventory, the RPS would initiate the mitigation actions. The pipe rupture would lead to decrease in reactor water level. When reactor water reaches Level-3 the reactor is shut down by scram, and when water level drops to Level-2, the RCIC auxiliary feedwater mode is initiated, to maintain and restore the reactor coolant inventory. If the water level further drops to Level-1.5, the high pressure core cooling provisions of the ECCS, the HPCF and RCIC, are automatically initiated to restore the coolant inventory if not already operating. Even if the coolant inventory could not be restored by those systems, the low pressure core cooling provisions of the ECCS, RHR LPFL mode with depressurisation by the ADS, are automatically initiated to mitigate the fault.

(b) Containment of Reactor Coolant in FDW Pipework beyond the RCPB

The steam generated in the RPV is transferred to the turbine side for power generation, condensed and supplied back to the RPV through the feedwater lines to close the cycle. The FDW subsystem of the NB consists of two FDW pipes, which contain reactor coolant (feedwater) during start-up, power generation and shutdown. In this case, the range of the RCPB goes up to the outboard check valve shown on Figure 12.3-12. The FDW piping beyond that valve is outside the RCPB range, but it still contains coolant that can be lost due to pipe rupture because of a mechanical failure or the effects of a hazard. Loss of reactor coolant would lead to a decrease in reactor coolant inventory and an increase of radioactive material release into the MS tunnel room.

In order to prevent a fault, the FDW is provided with instrumentation that controls the process parameters and informs the operators in the MCR about the plant state before any abnormality leads to a fault such as pipe rupture. Should damage or a pipe rupture occur, the instrumentation and control provisions would detect it promptly so that the corresponding mitigation measures can be put in place.

A rupture of the FDW pipe would cause an increase of radioactive material release by the coolant into the MS tunnel room, which has the same consequences as a Main Steam Line Break (MSLB) (bounding fault). The consequences can be mitigated by the check valves in the FDW line, which promptly stop the loss of reactor water through the pipe break. In addition, the Feedwater Pumps are promptly tripped to protect them when loss of pressure in the discharge side is detected.

If a rupture of the FDW pipe outside the RCPB should occur, the rupture of a FDW pipe would cause a decrease in reactor coolant inventory, which has the same consequences of a loss of all feedwater flow (bounding fault). The mitigation measures implemented are similar to a MSLB. When reactor water reaches Level-3 the reactor is shut down by scram initiated by the RPS, and when water Level-2 is reached the RCIC auxiliary feedwater mode is initiated to maintain and restore the reactor coolant inventory. If the reactor water level decreases further, the same actions as for the MSLB are taken.

(c) Long-term Heat Removal for Reactor Shutdown

The NB is provided with a total of 16 SRVs mounted on the MS lines between the RPV and the inboard MSIV within the D/W. Among them, any two of the seven SRVs provided with ADS function can be manually operated from the MCR to discharge the steam generated due to the decay heat in the RPV into the S/P for long-term heat removal in order to bring the reactor to the cold shutdown state in the event the Main Condenser was unavailable as heat sink. This operation is done in conjunction with the RCIC and the RHR Shutdown Cooling Mode.

By manually actuating the SRVs with ADS function during hot shutdown, RPV depressurisation is started so that the steam generated by the decay heat can be blow down into the S/P instead of the Main Condenser and thus reach reactor cold shutdown. The SRVs operate in conjunction with the RCIC, which should be already operating in the auxiliary feedwater mode after automatic initiation upon reaching Level 2 to supply makeup to the RPV instead of the unavailable FDW supply. The turbine-driven pump assembly of the RCIC will supply makeup from either the CST or the S/P to the RPV through FDW line B. The turbine will be driven with a portion of the decay heat steam from the RPV, and will exhaust to the S/P until the RPV is depressurised to a level heat removal can be continued by operation of the RHR Shutdown Cooling mode.

During reactor shutdown, in the event the RHR Shutdown Cooling operation was lost, the SRVs are manually actuated the same way to discharge the heat into the S/P and deliver long-term heat removal in conjunction with the reactor core cooling systems available.

The SRVs are connected to the HPIN which constantly supplies the SRVs with nitrogen gas from the nitrogen gas supply machine at a supply pressure sufficient to actuate and open them. The SRVs forming the ADS are provided with three three-way solenoid valves, which can be remotely and independently operated from the MCR to control the flow of nitrogen gas to the actuator of the SRVs.

These SRVs are also provided with one accumulator of 220L of capacity dedicated for ADS operation, which is kept charged by the HPIN nitrogen gas supply. Loss of the HPIN supply does not prevent opening of the SRVs because of the nitrogen gas stored in the accumulators. The capacity of the accumulator and the leakage rate of the SRV assembly are adequately designed to open and keep open the target SRVs under the pressure conditions in the PCV and RPV when the delivery of this function is required for the time required.

(d) Limitation of the Loss of Reactor Coolant till MSIV Closure

One MS Flow Restrictor is provided on each of the four MS lines to limit the loss of coolant and the release of radioactive material from the RPV following a MS line rupture outside the PCV. This is necessary to prevent RPV water level from dropping below the top of active fuel within the time required to close the MSIVs.

The MS Flow Restrictors are capable of limiting the main steam flow through the damaged pipes up to a maximum of 200% of the rated flow through each pipe thanks to their Venturi throat diameter.

(e) Isolation of the MS Lines under Pipe Rupture

The NB is provided with two MSIVs mounted in series on each MS line, one inside and the other outside the PCV, in order to limit the release of reactor coolant and radioactive material to the surroundings in the event of a MS pipe rupture.

If a MS line break occurs inside the containment, closure of the isolation valve outside the PCV seals the containment itself.

The MSIVs automatically close when they receive one of the isolation signals, which indicate MS pipe damage or rupture. In addition, remote manual switches in the MCR enable the operator to manually operate the valves if necessary.

The MSIVs are capable of closing in less than 4.5 seconds in order to control the release of reactor coolant and radioactive material in the event of a MS pipe rupture. At the same time, the MSIVs close slowly enough to ensure that simultaneous closure of all lines does not lead to an excessive pressure increase in the RCPB in the event of transients demanding closure of the valve.

Pneumatic supply for valves actuation is constantly supplied to the valves from the HPIN or the Instrument Air system (IA) at a supply pressure sufficient to actuate them. Inboard MSIVs are supplied nitrogen gas from the HPIN and outboard MSIVs are supplied air from the IA. This is to ensure maintaining an inert atmosphere within the PCV but also provides some diversity of supply.

The MSIVs are provided with one accumulator of 170L of capacity which is kept charged by the HPIN or the IA constant supply. This accumulator works as a backup of the operating gas and has sufficient capacity to store the necessary energy to close the valves even if the constant supply from the HPIN or the IA is lost.

(f) Maintenance of PCV Integrity when SRV Discharge

The steam discharged by the SRVs is transferred through the discharge piping to the quenchers below the S/P water surface to condense it. The NB is provided with one quencher (diffuser) at the end of each SRV discharge line to diffuse the main steam discharged by the SRVs inside the S/P so that this condenses in a uniform and stable manner and thus suppress the dynamic loads generated during discharge to ensure the integrity of the PCV.

In order to suppress the dynamic loads, the shape and dimensions of the quenchers are designed to minimise the void pulsation loads generated and to prevent local temperature peaks inside the S/P during the discharge of steam into it while operation of the SRVs.

(2) Equipment Design and Operation

(a) MS Flow Restrictors

(i) Purpose

- Limitation of the Loss of Reactor Coolant till MSIV Closure ([NB SFC 4-7.1])
  The MS Flow Restrictor limits the coolant blow-down rate from the RPV, in order to limit the loss of coolant and the release of radioactive material from the RPV in the event that a MS pipe break occurs outside the PCV. In addition, the MS Flow Restrictor is used to measure the mean steam flow for the signals to initiate closure of MSIVs.
- Reactor Coolant Pressure Boundary ([NB SFC 4-1.1])
  The MS Flow Restrictor as a part of the RCPB contains reactor coolant and forms a pressure barrier to prevent the loss of reactor coolant.

(ii) Configuration and Operation

The MS Flow Restrictors are Venturi type nozzles integrated into the steam outlet nozzle extension of the RPV and without dynamic parts. They are considered an integral part of the RPV. Limitation of the extent of reactor coolant blow-down is achieved by the geometry of the venturi structure.

In addition to the function to limit the loss of reactor coolant in the event of a MS pipe break, the flow restrictors are also used to measure the steam flow to generate the signals to initiate closure of the MSIVs when the steam flow exceeds the preselected operational limits. This is done by using the RPV dome pressure and the venturi throat pressure as the high and low pressure sensing locations for differential pressure measurements by the instrumentation provided in the flow restrictor.

(iii) Performance

In order to limit the loss of coolant for the delivery of [NB SFC 4-7.1], the MS Flow Restrictors are designed with a Venturi throat diameter that limits the coolant blowdown rate from the RPV to a flow rate less than or equal to 200% of the rated steam flow per MS pipe in the event that a MS pipe break occurs outside the PCV.

(b) MSIVs

(i) Purpose

- Isolation of the MS Lines under Pipe Rupture ([NB SFC 4-7.2])
  The MSIVs automatically close to isolate the MS lines in the event a pipe break occurs to limit the loss of coolant and the uncontrolled release of radioactive materials from the nuclear system.
- Primary Containment Vessel Boundary Function ([NB SFC 4-7.3])
  The MSIV as a part of the PCV boundary forms a barrier to confine radioactive material within the containment and prevents its dispersion to the environment in the event of faults.
- Reactor Coolant Pressure Boundary ([NB SFC 4-1.1])
  The MSIV as a part of the RCPB contains reactor coolant and forms a pressure barrier to prevent the loss of reactor coolant.

(ii) Configuration and Operation

The two MSIVs are welded in series on each of the four MS lines (eight MSIVs in total) to close them and thus limit the release of reactor coolant and radioactive material in the event of a MS pipe rupture.

Each MSIV is a Y-pattern, globe valve which permits the inlet and outlet passages to be streamlined and thus minimise the pressure drop during normal steam flow. This configuration is such that normal steam flow tends to close the valve, and higher inlet pressure drag force tends to hold the valve closed.

Attached to the upper end of the stem is a pneumatic cylinder that opens and closes the valve.

The pneumatic control unit is attached to the pneumatic cylinder, which contains three solenoid pilot valves and pneumatic control valves that control the flow of nitrogen or air to the upper and lower compartments of the piston in the pneumatic cylinder.

The MSIV is operated by pneumatic pressure (nitrogen or gas) to open and by pneumatic pressure with the support of compressed springs to close.

Valve quick-closure is performed when nitrogen or air is admitted to the upper piston compartment. This will isolate the MS lines in the event of a LOCA or other events requiring containment or steam line isolation to limit the release of reactor coolant.

The pneumatic cylinder provides the exclusive motive force for opening and the primary force for closing the MSIV.

(iii) Performance

The MSIVs are designed to quickly and automatically close within the time and conditions required by pneumatic actuation with the support of the springs for the delivery of [NB SFC 4-7.2]. The Y-pattern globe valve design is such that it allows the MSIVs to close within this time under the conditions of high pressure differential and fluid flow after a MS pipe break. Hence, the MSIV limits nuclear reactor coolant leakage and prevents damage to the reactor core if a large quantity of steam was released out of the system due to a rupture accident, etc. on the MS lines outside the PCV. At the same time, the MSIVs close sufficiently slowly so that simultaneous closure of all MS lines does not lead to excessive pressure increase in the RCPB in the event of transients.

(c) SRVs

(i) Purpose

- Long-term Heat Removal ([NB SFC 3-1.1], [NB SFC 3-1.2])
  The SRVs contribute to the delivery of long-term residual heat removal by discharging steam generated by the residual decay heat into the S/P.
- Overpressure Protection ([NB SFC 4-2.1], [NB SFC 4-6.1])
  The SRVs passively open by spring actuation in order to deliver RCPB overpressure protection function in the safety valve operation mode. Alternatively, the SRVs automatically open by powered actuation in order to deliver RCPB overpressure protection function in the relief valve operation mode.
- Re-seat Function ([NB SFC 4-5.1])
  The SRVs passively close when the steam inlet pressure reaches the re-seat set pressure in order to prevent excessive loss of reactor coolant when operating as safety valves by spring actuation.
- Reactor Core Cooling ([NB SFC 2-1.1], [NB SFC 2-1.2], [NB SFC 2-1.3], [NB SFC 2-1.4], [RDCF SFC 2-2.1], [RDCF SFC2-2.2], [RDCF SFC2-2.3], [RDCF SFC2-2.4], [RDCF SFC2-2.5], [RDCF SFC 2-2.6])
  The SRVs contribute to the delivery of reactor core cooling by maintaining the necessary pressure conditions for high pressure core cooling systems function. In addition, the SRVs are used to depressurise the RPV to a level where the lower pressure core cooling systems can deliver water injection into the RPV.
- Reactor Coolant Pressure Boundary ([NB SFC 4-1.1])
  The SRV, as a part of the RCPB, contains reactor coolant and forms a pressure barrier to prevent the loss of reactor coolant.

(ii) Configuration and Operation

The SRVs are spring-loaded safety valves to which an actuator is attached to remotely force them to open and close by supplying nitrogen gas to the piston of the actuator.

For the delivery of reactor depressurisation for core cooling or long-term heat removal by the ADS or the RDCF, the nitrogen gas is supplied by the accumulator dedicated for these functions (ADS Accumulator or RDCF Accumulator), which are kept charged in plant normal conditions by the HPIN. The SRV is opened by energising any of the three dedicated solenoid pilot valves in the case of the ADS or any of the two dedicated solenoid pilot valves in the case of the RDCF. When energised by DC power, the solenoid valves switch to the supply port opening the path to discharge the nitrogen from the accumulator into the pneumatic cylinder of the actuator. Once the solenoid valves are de-energised, they switch to the exhaust port discharging the nitrogen gas from the cylinder of the actuator so that the valve returns to the normal closed position. The solenoid pilot valves form an integral part of the SRV assembly.

For the delivery of core cooling for long-term (beyond 24hrs) or when Class 2 support systems (control or power) are not available such as during Station Blackout (SBO) with Common Cause Failure (CCF) of EDG and Back-up Building Generator System (BBG), the nitrogen gas supply comes from the RDCF dedicated nitrogen gas cylinders. The SRV is actuated by the switching valve, which opens due to the nitrogen gas pressure pushing at the valve seat, which raises it opening the nitrogen gas flow path to the cylinder actuator of the SRV. The switching valve remains opened as long as the nitrogen gas supply is maintained. To close the valves, and consequently the SRVs, the nitrogen gas supply from the cylinders has to be discharged so that the pressure on the switching valve seat drops, allowing the seat to return to the close position and thereby opening the exhaust port for the nitrogen gas in the SRV cylinder actuator. The switching valve is a three-port shuttle valve that is regarded as a supplement of the SRV just like the solenoid valves, and therefore, forms an integral part of the SRV assembly.

For the safety valve operation mode, the valves passively open when the main steam pressure at the valve inlet reaches the set lift pressure of the spring of the associated valve. The valve keeps open while the steam pressure is above the re-seat pressure. At this point the spring force overcomes the steam pressure and the valve re-closes.

For the relief valve operation mode, the nitrogen gas is supplied by the accumulator dedicated for the relief function (SRV Relief Accumulator), which is kept charged in plant normal operation condition by the HPIN. The SRV is opened by energising the dedicated three-way solenoid pilot valve, which if energised by DC power, opens the path to discharge nitrogen into the pneumatic cylinder of the actuator. Once the solenoid valve is de-energised, this switches to exhaust the nitrogen gas from the actuator so that the valve returns to the normal closed position.

(iii) Performance

- <u>Overpressure Protection ([NB SFC 4-2.1], [NB SFC 4-6.1])</u>
  Sizing of the SRV capacity is based on establishing an adequate margin from the peak RCPB pressure to the vessel code limits in response to transients and accidents. Whenever the system pressure increases to the valve spring set pressure of a group of valves, these valves are assumed to begin opening and to reach full open at 103% of the valve spring set pressure.
  The SRV spring force is set such that the setting lift pressure and discharge capacity meet the requirements of Table 12.3-4 in section 12.3.3.3 ‘RCPB Overpressure Protection System’ for each group of valves when delivering the safety valve operation mode.
  The SRV is designed with a main disk full stroke opening time below the specified limit when they are spring-actuated for the delivery of the safety valve operation mode.
  Each group of SRVs is designed such that the valves open and deliver the required discharge capacity at the set points specified in Table 12.3-5 in section 12.3.3.3 ‘RCPB Overpressure Protection System’ when automatically actuated by nitrogen gas for the delivery of the safety valve operation mode.

- <u>Re-seat Function ([NB SFC 4-5.1])</u>
  The internal structure of the SRVs is designed such that the valves passively re-close by the spring force when the steam reaches a pressure at the valve inlet no less than the minimum percentage of the spring lifting set pressure when delivering the safety valve operation mode.

- <u>Long-term Heat Removal ([NB SFC 3-1.1], [NB SFC 3-1.2])</u>
  Sizing of the SRVs does not depend on the system requirements for the delivery of long-term heat removal. The bounding conditions for sizing of the SRVs are the requirements from the reactor overpressure protection function. Based on this, since the SRVs have a minimum discharge capacity of 460t/h each, operation of a maximum of two of them (one valve could be sufficient

depending on the conditions) is enough to perform reactor blow-down for long-term heat removal depending on the conditions.
With regard to the leakage, the SRV assembly overall leakage rate is set sufficiently low so that maintenance of the valve at the open position for at least 24 hours without any other external support than the nitrogen gas stored in the accumulator is possible under the required pressure conditions

- <u>Reactor Core Cooling ([NB SFC 2-1.1], [NB SFC 2-1.2], [NB SFC 2-1.3], [NB SFC 2-1.4], [RDCF SFC 2-2.1], [RDCF SFC2-2.2], [RDCF SFC2-2.3], [RDCF SFC2-2.4], [RDCF SFC2-2.5], [RDCF SFC 2-2.6])</u>
  Sizing of the SRVs does not depend on the system requirements for the delivery of this function. The bounding conditions for sizing of the SRVs are the requirements from the reactor overpressure protection function. Based on this, since the SRVs have a minimum discharge capacity of 460t/h each, operation of a maximum of seven of them (fewer valves could be sufficient depending on the reactor conditions) is enough to perform reactor blow-down for reactor core cooling.
  With regard to the leakage, the SRV assembly overall leakage rate is set sufficiently low so that maintenance of the valve at the open position for at least 24 hours without any other external support than the nitrogen gas stored in the accumulator is possible under the required pressure conditions.

(d) SRV Discharge Line Quencher

(i) Purpose
<u>Maintenance of PCV Integrity when SRV Discharge ([NB SFC 4-7.4])</u>
The SRV discharge line quenchers diffuse the main steam discharged by the SRVs inside the S/P so that this condenses in a uniform and stable manner and thus suppress the dynamic loads generated during discharge to ensure the integrity of the PCV.

(ii) Configuration and Operation
The SRV discharge line quencher is a X-type diffuser device consisting of a short conical extension of the bottom end of the SRV discharge line and a capped cylindrical central section (plenum), from which four capped arms perforated with many small holes extend. The arms direct the air and steam mixture over a broad area inside the S/P, and thus support heat transfer during SRV actuation between the hot compressed steam and air mixture and the colder water in the S/P. There are 16 quenchers in total, one for each SRV.

(iii) Performance
The distance from S/P floor to the centre line of the SRV discharge line quencher arm, the number of arms, the length and orientation of the arms, the number of holes and the diameter of the holes have been set such that the current design ensures minimisation of void pulsation loads generated during discharge of steam as well as prevention of local peak temperatures in the S/P for the delivery of [NB SFC 4-7.4].

(3) Main Support Systems

(a) Instrumentation and Control Systems

Instrumentation and control are provided to measure and monitor the operating conditions of the NB components necessary for the delivery of the safety functions claimed. The relevant instrumentation and control provisions are described as follows:

(i) Containment of Reactor Coolant beyond the RCPB
The following items are monitored in order to detect damage or rupture in the MS and FDW piping outside the RCPB:

- Reactor Water Level (RVI),
- R/B Waste Sump Discharge Flow (LDS),
- Temperature in the MS Tunnel Room within the R/B (LDS),
- MS Flow (RVI), and
- MS Pressure (RVI).

(ii) Long-term Heat Removal for Reactor Shutdown
Manual operation of the SRVs for long-term heat removal function for reactor shutdown is controlled by the logic of the SSLC and operation is performed through the MCR HMI.
The main parameters measured in relation to the delivery of this function are the position of the SRVs, the temperature of the SRV discharge line and the nitrogen supply pressure to the valves, which are all displayed in the MCR.

(iii) Isolation of the MS Lines under Pipe Rupture
Automatic isolation of the MS lines by the MSIVs is controlled by the logic of the SSLC, which governs the PCIS that gives the order to close the valves.
The status of the MSIVs can be controlled through the MCR HMI, which also provides switches for manual isolation if necessary.
The MSIV automatic closure signals under pipe rupture are the following:

- Reactor Water Level Low (water level 1.5),
- Main Steam Line Flow High,
- Main Steam Line Tunnel Temperature High, and
- Main Steam Line Pressure Low.

The design of the instrumentation and control systems is justified in Chapter 14: Control and Instrumentation.

(b) Power Supply System

(i) Containment of Reactor Coolant beyond the RCPB
The instrumentation and controllers to control the process parameters of the MS and FDW piping outside the RCPB and detect their failure are supplied power by different divisions of the Class 1 AC instrumentation power supply system.

(ii) Long-term Heat Removal for Reactor Shutdown
The three three-way solenoid valves for manual actuation of the SRVs through the ADS Accumulator are supplied power by DC Division I, Division II and Division III of Class 1 DC Power Supply System.

(iii) Isolation of the MS Lines under Pipe Rupture
The two solenoid pilot valves for MSIV closure are supplied power by two functionally independent divisions of Class 1 AC UPS supported by the battery backed Class 1 DC power supply system. The MSIVs are fail-safe valves that close in the event of loss of both power supplies.

The design of the power supply system is justified in Chapter 15: Electrical Power Supplies.

Assumptions, Limits and Conditions for Operation

In order to ensure that the NB is operated within the safety limits and that the design requirements from the safety case are met during the operating regime, appropriate LCO and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the ‘Basis of Safety Cases on Nuclear Boiler System’ (Ref 12.3-4).

- Each MSIV shall be operable during start-up, power operation and hot shutdown for the delivery of the SFCs claimed when required.
- The ADS function of the seven SRVs shall be operable during start-up, power operation and hot shutdown for the delivery of the SFCs claimed when required.

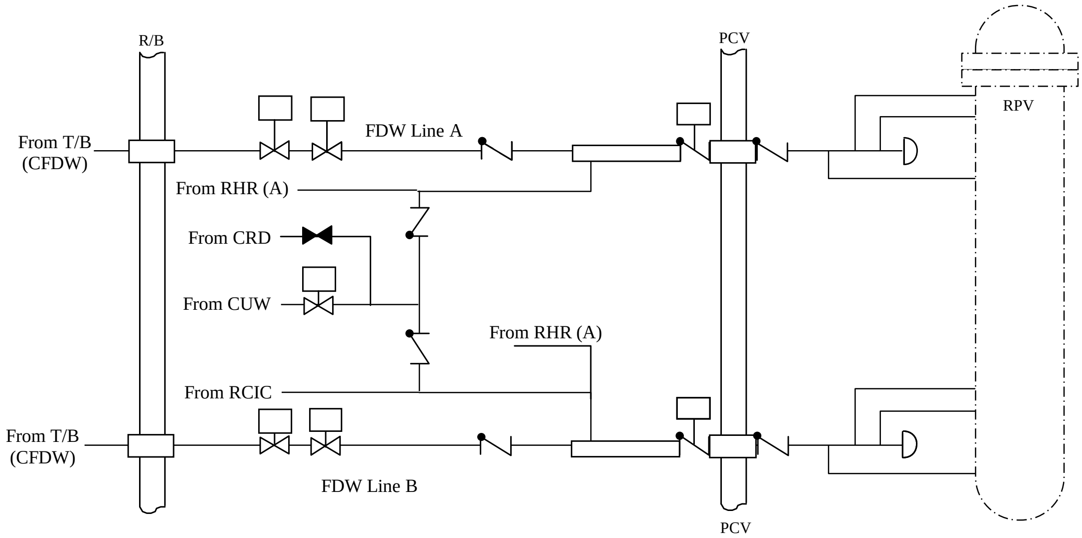

Figure 12.3-12 : Outline of the FDW

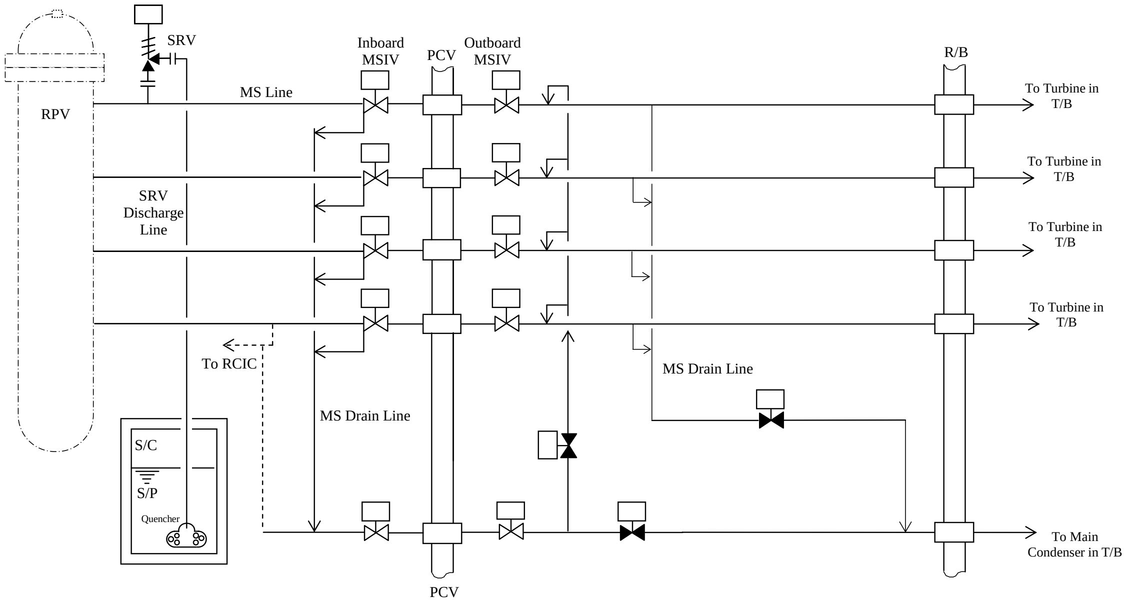

Figure 12.3-13 : Outline of the MS (R/B portion)

12.3.5.3 Reactor Water Clean-up System (CUW)

System Summary Description

This section is a general introduction to the CUW where the system roles, system functions, system configuration and modes of operation are briefly described. The CUW safety case is justified in the 'Basis of Safety Cases on Reactor Water Clean-up System' (Ref 12.3-5). The CUW design is described in detail in the system design specifications (Ref 12.3-16) and the P&IDs (Ref 12.3-17).

Further information regarding management of the Primary Reactor Coolant Water Chemistry is available in Chapter 23: Reactor Chemistry, Section 23.3.

(1) System Roles

The CUW removes impurities contained in reactor water and maintains the quality of reactor water within the determined range in order to mitigate the following:

(a) Corrosion of the equipment and piping making up the reactor coolant system.

(b) Decrease of heat transfer efficiency by adhesion of impurities on the fuel surface.

(c) Radioactive contamination of the reactor coolant system and the related equipment.

(2) Functions Delivered

The CUW is designed to perform the following functions:

(a) The CUW provides a continuous purifying treatment of reactor water by removing soluble and insoluble impurities.

(b) The CUW provides a route to discharge the excess volume of reactor water into the LCW Collection Tank or the S/P during start-up when temperature increase causes expansion of water in the reactor circuit.

(c) The CUW recirculates reactor water by taking suction of water from the bottom of the RPV ensuring a flow around the CRD nozzles at the bottom of the RPV lower plenum.

(d) The CUW cools the RPV by spraying water via the RPV spray head, when the replacement of fuel is required to be initiated early during a refuelling outage.

(3) Basic Configuration

The CUW consists of the following components:

(a)	CUW Regenerative Heat Exchanger	1 unit
(b)	CUW Non-Regenerative Heat Exchanger	2 units
(c)	CUW Pump	2 units
(d)	CUW Filter Demineralizer	2 units
(e)	Piping and Valves	1 set
(f)	Instruments and Control Components	1 set

Figure 12.3-14 shows an outline of the CUW.

(4) Modes of Operation

The CUW can deliver the following operation modes by switching the position of the valves.

(a) Reactor Water Clean-up Operation Mode (Normal Operating Mode)

During normal conditions, reactor water is drawn from the RPV drain line and the RHR division (B) suction line and transferred through the Regenerative Heat Exchanger (tube side) and the Non-Regenerative Heat Exchangers (tube side). Then, reactor water is pressurised by the CUW Pumps, cleaned up by the CUW Filter Demineralizers, and passed through the shell side of the Regenerative Heat Exchanger. Finally, reactor water is returned into the RPV through the FDW lines.

(b) Start-up Mode

During reactor start-up, the water flow from the CRD and the expansion in the volume of reactor water (caused by the maximum temperature increases of 55°C /h) are discharged through the CUW into the LCW Collection Tank or the S/P so that reactor water is maintained at an appropriate level until steam is bypassed directly to the Main Condenser.

(c) Hot Shutdown Mode

(i) Reactor Water Level Maintenance Operation

The CUW automatically regulates the flow control valve and discharges the water provided by the CRD, the RIPs and the CUW Pump purge lines, into the LCW Collection Tank or the S/P so that the reactor water level is maintained at a Normal Water Level (NWL) during Hot Shutdown Mode.

(ii) Circulation Operation

When the RIPs are shut off, the CUW is initiated to circulate reactor water in order to minimize the temperature gradient in the RPV. The circulation routes are described as follows. Reactor water is drawn from the bottom of the RPV and the RHR division B, transferred through the Regenerative Heat Exchanger (tube side) and the Non-Regenerative Heat Exchangers, pressurized by the CUW Pumps, purified at the CUW Filter Demineralizer or bypassed, and then, through the Regenerative Heat Exchanger (shell side). Finally, water is returned into the RPV through the FDW lines.

(d) Refuelling Mode

During the refuelling operation, the CUW continuously cleans up reactor water in conjunction with the FPC. In addition, the CUW is capable of removing the spare water in the reactor above the normal reactor water level following refuelling. Moreover, the CUW is capable of bypassing the Regenerative Heat Exchanger in order to improve capacity of removing the decay heat when the RHR is shut off to prevent oscillations of the water surface in the RPV during refuelling outages.

(e) RPV Spray Mode

The CUW provides full system flow to the RPV head spray as required, for more rapid cooling of the RPV when a fuel replacement is required to be initiated early during a refuelling outage.

Design Bases

This section describes the design bases for the CUW. The CUW has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

Normal Conditions

(1) The CUW provides a continuous purifying treatment of reactor water by removing soluble and insoluble impurities during normal conditions. [CUW SFC 5-8.1]

(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

(2) The CUW piping and components outside the RCPB contain reactor coolant. A breach could lead to a release of radioactive material of dose consequences that are relatively low, but demanding Safety Category A safety functions to mitigate them. [CUW SFC 4-3.1]

(This function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements)

Normal and Fault Conditions

(3) The CUW portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [CUW SFC 4-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified as explained in section 12.3.3.1)

Fault Conditions

(4) The CUW components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults. [CUW SFC 4-7.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in Generic PCSR section 13.3.3.2 ’Primary Containment Isolation System’)

System Design Description

This section describes the design of the CUW to support and justify the delivery of [CUW SFC 5-8.1]. Additional design description can be found in Ref 12.3-5, Ref 12.3-16 and Ref 12.3-17.

(1) Overall System Design and Operation

The CUW is designed to operate continuously at 100% of capacity during normal conditions. In addition, the CUW is capable of operating at 50% of capacity with either train of the CUW Pumps and Non-Regenerative Heat Exchangers.

For reactor water clean-up mode during normal conditions, reactor water is taken from the suction line which forms part of division B of the RHR and from the drain line at the bottom of the RPV. Reactor water is pressurised, circulated by the CUW Pumps and cooled down to the specified temperature through the Regenerative Heat Exchanger (tube side) and the Non-Regenerative Heat Exchangers (tube side).

Soluble and insoluble impurities contained in reactor water are continuously removed through the ion exchange resin in the CUW Filter Demineralizers. Reactor water is reheated by the Regenerative Heat Exchanger (shell side) after demineralization and returned into the RPV through the FDW line.

(2) Equipment Design and Operation

(a) CUW Pump

(i) Purpose

The CUW Pumps provide water flow from the reactor vessel through heat exchangers and filter demineralizers for clean-up, and return it to the reactor vessel in order to deliver [CUW SFC 5-8.1].

(ii) Configuration and Operation

The main configuration aspects with regard to the CUW Pump are described as follows.

- The CUW Pump is designed so that it can maintain a sufficient flow rate during normal conditions.
- Two 50% capacity CUW Pumps are mounted in parallel and normally both of the pumps are operated.
- Each pump is provided with a casing filling water line to prevent contamination of the piping between the pump and the stop valve for maintenance and to perform water filling.

(iii) Performance

The specification of the CUW pump required to ensure the delivery of [CUW SFC 5-8.1] is shown below.

Number:	2 units
Pump Type:	Seal-less centrifugal Pump
Capacity:	approx. 77 m^3/h/unit

(b) CUW Heat Exchanger

(i) Purpose

The CUW Heat Exchangers include one Regenerative Heat Exchanger and two Non-Regenerative Heat Exchangers. This equipment is designed to cool the reactor water for protecting the CUW Filter Demineralizers in order to deliver [CUW SFC 5-8.1].

(ii) Configuration and Operation

The main configuration aspects with regard to the CUW Heat Exchangers are described as follows.

- The Regenerative and Non-Regenerative Heat Exchangers are installed in series. The system is provided with one 100 %-capacity Regenerative Heat Exchanger and two 50 %-capacity Non-Regenerative Heat Exchangers installed in parallel (both units are operated during Normal Operation Mode).
- The outlet temperature on the tube side of the Non-Regenerative Heat Exchangers is controlled at 49°C during Normal Operation Mode.
- The outlet temperature on the tube side of the Non-Regenerative Heat Exchangers does not exceed 60°C during all operating modes. If the temperature exceeds 60°C, the isolation valves are designed to be closed by the temperature switch and the CUW Pumps are shut down. These operations are performed to protect the resin installed in the CUW Filter Demineralizer against high temperatures.
- The Non-Regenerative Heat Exchangers are cooled by the RCW.
- The Non-Regenerative Heat Exchangers are designed to be separated into two trains so that leakage from one heat exchanger can be isolated and prevent discharge of reactor water from the tube side of heat exchanger to outside of the system.
- Leak detection equipment is provided to detect leakage from the flange seat (between double gaskets) of the Non-Regenerative Heat Exchanger tube side.

(iii) Performance

The specification of the CUW Regenerative Heat Exchanger required to ensure the delivery of [CUW SFC 5-8.1] is shown below:

Number: 1 unit
Type: Horizontal U-tube type
Design Conditions:

	Tube side	Shell side
Fluid:	Reactor water	Cleaned reactor water
Flow:	152.5×10^3 kg/h	152.5×10^3 kg/h
Heat exchange capacity:	31.9 MW/unit	

The specification of the CUW Non-Regenerative Heat Exchanger required to ensure the delivery of [CUW SFC 5-8.1] is shown below:
Number: 2 units
Type: Horizontal U-tube type
Design Conditions:

	Tube side	Shell side
Fluid:	Reactor water	RCW water
Flow:	76.25×10^3 kg/h	160×10^3 kg/h
Heat exchange capacity:	5.58 MW/unit	

(c) CUW Filter Demineralizer

(i) Purpose
The purpose of the CUW filter demineralizer is to clean-up the coolant water from impurities and contain them to avoid their dispersion in the environment in order to deliver [CUW SFC 5-8.1] and [CUW SFC 4-1.1].

(ii) Configuration and Operation
The CUW Filter Demineralizer equipment consists of filter demineralizer elements, holding pumps, piping, valves, and auxiliary equipment for pre-coating/backwashing/disposal of the resin, instruments and controllers.
Two 50%-capacity filter demineralizer elements are installed in parallel and continuously operated during normal conditions.
The filter demineralizer elements are designed based on Normal Operating Mode as standard.
A flow control device is installed on the filter demineralizer element in order to regularly maintain the flow rate through both lines.

(3) Main Support Systems

(a) Instrumentation and Control Systems

The main instrumentation and control provisions related to CUW operation from the performance and reliability points of view are summarised as follows.

(i) Instrumentation

- Flow meters are provided to the following lines:
  - CUW Pump suction line (system inlet),
  - Return line to the reactor (system outlet), and
  - CUW Pump discharge line (one flow element is arranged on each pump)
    The system flow rate is controlled by the flow control valve mounted on the CUW Filter Demineralizer. The flow rate of the RCW (shell side of the Non-Regenerative Heat Exchangers) is regulated by the temperature control valve in order to maintain the outlet temperature at the determined value.
- The differential pressure across the Filter Demineralizer is indicated on the local panel.
- The reactor water conductivity at the inlet and outlet of the CUW Filter Demineralizer is continuously monitored and recorded in the MCR for Filter Demineralizer backwashing and pre-coating. An alarm is initiated if the water quality at the inlet and the outlet of the Filter Demineralizer element is worse than the specified value.
- Devices are provided to measure the inlet/outlet temperature on the tube side and the outlet temperature on the shell side of the CUW Regenerative Heat Exchanger, and the temperature in the suction piping from the bottom of the RPV. The measured values are recorded and confirmed in the MCR.

(ii) Control

The main control provisions related to the delivery of the CUW safety functions are summarised as follows.

<u>System Isolation</u>

The isolation valves are closed automatically upon the following signals.

- Reactor water level — Low
- Temperature of the CUW room atmosphere — High
- Differential flow between system inlet/outlet — Large
- Outlet temperature of the Non-Regenerative Heat Exchanger — High
- ATWS conditions (related to Standby Liquid Control System (SLC) initiation)

Assumptions, Limits and Conditions for Operation

Based on the general principles for the identification of Assumptions and LCOs described in Standard Control Procedure for Identification and Registration of Assumptions, Limits and Conditions for Operation (Ref. 12.1-8 “XD-GD-0042”), limits directly related to operational aspects, ex. Operation Control, will be determined in the site specific phase and the details will commensurate with the maturity of the design. For this reason there are no applicable LCOs on this system within the scope of the GDA.

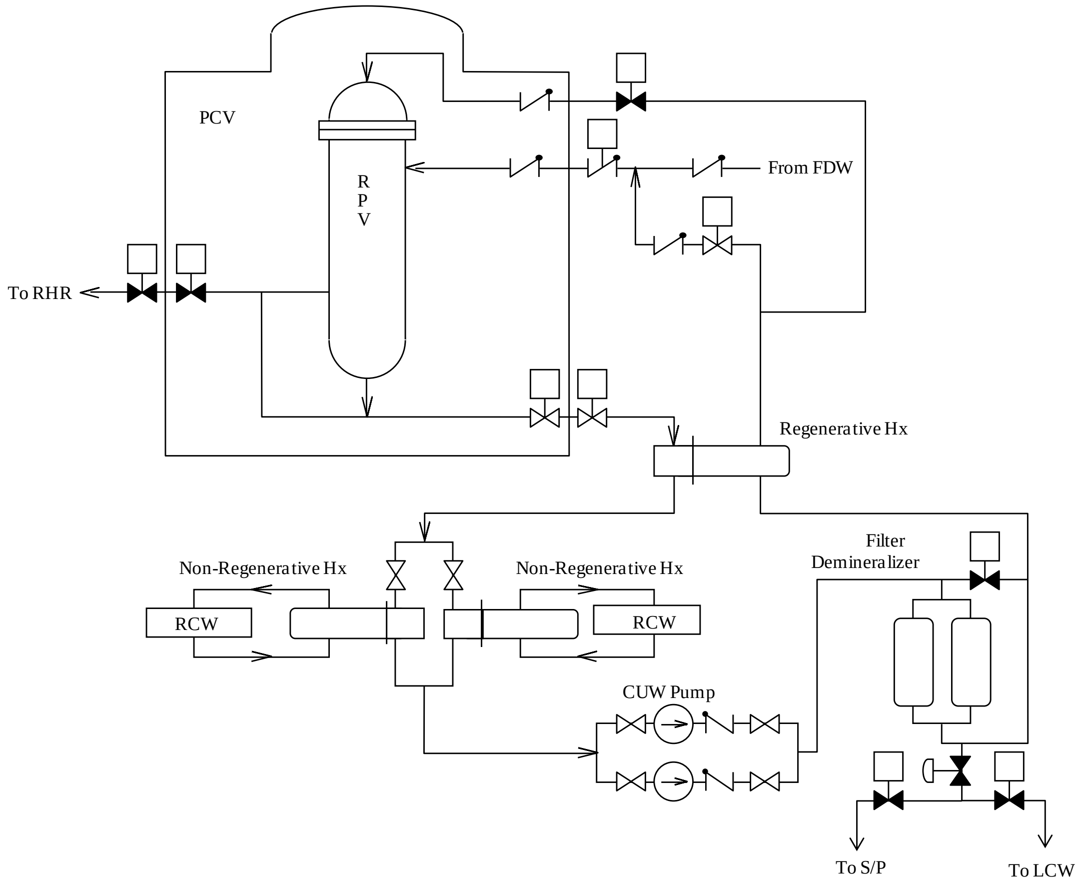

Figure 12.3-14 : Outline of the CUW

12.3.5.4 Residual Heat Removal System (RHR)

System Summary Description

This section is a general introduction to the RHR where the system roles, system functions, system configuration and modes of operation are briefly described. The RHR safety case is justified in the 'Basis of Safety Cases on Residual Heat Removal System' (Ref 12.3-6). The RHR design is described in detail in the system design specifications (Ref 12.3-18) and the P&IDs (Ref 12.3-19, Ref 12.3-20 and Ref 12.3-21).

(1) System Roles

The main roles of the RHR are to remove decay heat during normal reactor shutdown and in the event of unavailability of the main condenser, and to cool the reactor core in the event of a LOCA.

(2) Functions Delivered

The RHR is designed to perform the following main functions:

(a) The RHR provides cooling to remove decay and sensible heat from the reactor after normal shutdown and in the event that the main condenser is not available.

(b) The RHR provides core cooling water supply to the reactor to compensate for water loss in the event of LOCA via the LPFL mode as part of the ECCS.

(c) The RHR removes heat from the PCV by cooling the water of the S/P.

(d) The RHR provides PCV cooling through sprays provided in the D/W and the Suppression Chamber (S/C) to remove heat and condense steam in the containment following a LOCA and thus prevent over pressurisation of the PCV. In addition, the drywell sprays can remove fission products from the containment atmosphere.

(e) The RHR can work as a backup to cool the Spent Fuel Pool (SFP) if the heat load exceeds the FPC maximum cooling capacity (e.g. during a full core off-load to the SFP).

(f) The RHR transfers S/P water to the Suppression Pool Water Drainage System (SPD) surge tank or the Liquid Waste System prior to the S/P maintenance.

(3) Basic Configuration

The RHR consists of three independent divisions, A, B, and C. Each division has water injection function into the RPV and heat removal function from the RPV or the PCV. The necessary piping, valves, pumps and heat exchangers are included in each division.

The RHR includes all the process pipes, strainers, pumps, motors, valves, instrumentation and controllers as shown on Figure 12.3-15. The main components are summarised as follows:

(a) Pumps, piping, heat exchangers and valves included in the discharge lines running from the pumps to the S/P, feedwater spargers (to the connection with FDW), drywell/wetwell spray spargers, and water injection spargers (LPFL) into the RPV.

(b) Valves, piping, and strainers included in suction lines from the S/P to the respective pumps.

(c) Valves and piping included in suction lines from the reactor shutdown cooling suction nozzle to the connection to the S/P suction line.

(d) Minimum flow piping and test piping with their valves.

(e) Instrumentation, controllers, operation logic/circuit and control panels.

(f) Piping and valves connecting to the FPC.

(4) Modes of Operation

The RHR can deliver the following operation modes by switching the position of the valves.

(a) Reactor Shutdown Cooling (SDC) Mode

The RHR provides reactor cooling to maintain cold shutdown during reactor normal shutdown and in the event that the main condenser is not available. The RHR removes the reactor decay heat so that refuelling and maintenance can be implemented after shutdown. The three RHR divisions are provided with this mode Figure 12.3-16. illustrates the RHR configuration for this mode of operation.

(b) LPFL Mode

The RHR cools the reactor core during LOCA (removal of decay heat from the reactor core) as part of the ECCS. The LPFL operates to cool the core in conjunction with the HPCF, the RCIC and the ADS in order to maintain the fuel cladding temperature below the design basis criteria. The three RHR divisions are provided with this mode. Figure 12.3-17 illustrates the RHR configuration for this mode of operation.

(c) PCV Spray Cooling Mode

The RHR removes the heat and condenses steam inside the drywell and wetwell after a LOCA in order to prevent overpressurisation of the containment. In addition, it removes fission products from the PCV atmosphere. RHR divisions B and C are provided with this mode. Figure 12.3-18 illustrates the RHR configuration for this mode of operation.

(d) Suppression Pool Cooling Mode

During this mode the RHR cools the S/P to remove the heat released as required. The three RHR divisions are provided with this mode. Figure 12.3-19 illustrates the RHR configuration for this mode of operation.

(e) Auxiliary Operation Mode

In addition, all three RHR divisions are provided with the following auxiliary operation modes:

(i) Fuel Pool Cooling Function
The RHR removes the residual heat from the spent fuel to support the FPC when the core is fully offloaded. By switching position of the valves, RHR division A and B are aligned with divisions A and B of the FPC and circulate SFP water through the heat exchangers. Figure 12.3-20 illustrates the RHR configuration for this mode of operation.

(ii) S/P Water Transfer Function
The RHR transfers S/P water to the SPD surge tank or the Liquid Waste System prior to the S/P maintenance.

Design Bases

This section describes the design bases for the RHR. The RHR has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown in Appendix A.

Normal Conditions

(1) The RHR through its Reactor Shutdown Cooling mode is the principal means to remove residual heat after normal reactor shutdown to reach reactor cold shutdown. [RHR SFC 3-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(2) The RHR provides the FPC with supplemental cooling to maintain the SFP water temperature within the design values by removing decay heat in the event of a full core offload where the heat load to the pool exceeds the FPC cooling capacity. This function can also be used for recovery from potential upper pools cooling failure and subsequent boiling event. [RHR SFC 2-4.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. It is developed and justified in the section related to the Spent Fuel Pool Cooling and Makeup Systems in Chapter 19: Fuel Storage and Handling)

(3) When the RHR is operating in Shutdown Cooling mode, its piping and components outside the RCPB contain reactor coolant. A breach could lead to a release of radioactive material of dose consequences relatively low, but demanding Safety Category A safety functions to mitigate them. [RHR SFC 4-3.1]

(This function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 3 requirements)

(4) When the RHR is in standby, its piping and components contain material with low radioactivity. A breach could lead to a release of radioactive material of dose consequences relatively low. [RHR SFC 4-4.1]

(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

Normal and Fault Conditions

(5) The RHR portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [RHR SFC 4-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified as explained in section 12.3.3.1)

Fault Conditions

(6) The RHR through its LPFL mode is a principal means to provide reactor core cooling as part of the ECCS when the RPV is in low pressure state so that significant damage to the fuel is prevented and the reaction between the fuel cladding and the reactor coolant is minimised in the event of infrequent faults such as LOCA. [RHR SFC 2-1.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in Chapter 13: Engineered Safety Features, Section 13.4 related to the ECCS)

(7) The RHR through its Reactor Shutdown Cooling mode is a principal means to deliver long term containment heat removal by removing the decay heat of fission products from the reactor without exceeding the fuel design margins and RCPB design conditions after reactor shutdown following frequent faults such as main condenser unavailability, and infrequent faults such as SBO after power recovery. [RHR SFC 3-1.2]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(8) The RHR through its Suppression Pool Cooling mode (SPC) is a principal means to deliver long term containment heat removal following frequent faults such as main condenser unavailability and infrequent faults such as Anticipated Transient Without Scram (ATWS). [RHR SFC 3-1.3]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. It is developed and justified in the section related to Containment Heat Removal in Chapter 13: Engineered Safety Features, Section 13.3.3.4 related to the Containment Heat Removal System)

(9) The RHR through its LPFL mode is a principal means to deliver long-term containment heat removal following frequent faults such as main condenser unavailability and infrequent faults such as LOCA. [RHR SFC 3-1.4]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. It is developed and justified in the section related to Containment Heat Removal in Chapter 13: Engineered Safety Features, Section 13.3.3.4 related to the Containment Heat Removal System)

(10) The RHR through its Suppression Pool Cooling mode (SPC) is a principal means to deliver long term containment heat removal upon RHR recovery following venting during infrequent faults such as SBO. [RHR SFC 3-1.5]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. It is developed and justified in the section related to Containment Heat Removal in Chapter 13: Engineered Safety Features, Section 13.3.3.4 related to the Containment Heat Removal System)

(11) The RHR components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults. [RHR SFC 4-7.1]

(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. It is developed in Chapter 13: Engineered Safety Features, Section 13.3.3.2 related to the PCIS)

(12) The PCV Spray Cooling mode of the RHR contributes to suppress PCV atmosphere pressure and remove fission products from the containment atmosphere during a LOCA inside PCV. [RHR SFC 4-7.2]

(This function is categorised as Safety Category B and the components to deliver it are designed to meet Safety Class 2 requirements. It is developed and justified in the section related to Containment Heat Removal in Chapter 13: Engineered Safety Features, Section 13.3.3.4 related to the Containment Heat Removal System)

NOT PROTECTIVELY MARKED

System Design Description

This section describes the design of the RHR to support and justify the delivery of [RHR SFC 3-1.1] and [RHR SFC 3-1.2]. Additional design description can be found in Ref 12.3-6, Ref 12.3-18, Ref 12.3-19, Ref 12.3-20 and Ref 12.3-21.

(1) System Design and Operation

The RHR is composed of three electrical and mechanical independent divisions designated A, B, and C. Each division contains the necessary piping, pumps, valves and heat exchangers.
By switching the position of the valves, the RHR can operate to deliver heat removal after reactor shutdown as described as follows.

(a) Standby Mode

The RHR is on standby and the motor-operated valves are at their normal positions (closed/open) during plant normal operation. The pump discharge lines are continuously kept filled with water during standby to avoid delay to providing cooling flow and to avoid pipe work and equipment stresses associated with sudden introduction of fluids. Relief valves mounted on the pump discharge pipelines protect them from overpressure.

(b) SDC

In the normal shutdown process, steam flow is reduced as the turbine load is reduced and once the turbine is shut down, the steam is blown to the main condenser via the turbine bypass valves to be cooled and condensed until the reactor pressure reaches 0.93MPa [gauge] or less, when operation of the Reactor Shutdown Cooling mode is possible. When reactor pressure is below 0.93MPa [gauge], and the reactor mode switch is in “shutdown” or “refuelling” position, the RHR Reactor Shutdown Cooling Mode is initiated, residual heat is removed from the reactor water at a rate below the RCPB cooling rate limit of 55°C/h to reduce the temperature to 52°C within 20 hours after CRs insertion. Finally, the water temperature is maintained or reduced to perform refuelling or service inspections.

(i) Under LOOP and single failure of one of the RHR divisions in the event the main condenser is unavailable, the RHR is capable of bringing the reactor to cold shutdown condition of 100°C within 36 hours following CRs insertion.

(ii) Reactor water is directly drawn from the RPV through the reactor shutdown cooling suction nozzle, passes through the RHR Heat Exchanger, is cooled and returned to the reactor prior to opening the RPV. Division A returns water through the feedwater line A, and divisions B and C return water through the respective low pressure flooder return lines.

(iii) This operation mode is initiated and stopped by operator's manual operation from the MCR, which would involve automatic and remote realignment of certain valves. The only operations performed outside the MCR for normal shutdown are manual operation of local flushing water admission valves, which are the means of providing clean water to the shutdown portions of the RHR.

(2) Equipment Design and Operation

(a) RHR Pump

(i) Purpose

The purpose of the RHR Pump is to supply water for the various modes of the RHR in order to deliver [RHR SFC 3-1.1], [RHR SFC 3-1.2], [RHR SFC 2-1.1], [RHR SFC 2-4.1], [RHR SFC 3-1.3], [RHR SFC 3-1.4], [RHR SFC 3-1.5] and [RHR SFC 4-7.2].

(ii) Configuration and Operation

Each division of the RHR is provided with one turbo type pump of approximately 954m^3/h design flow rate driven by an induction motor to deliver heat removal after reactor shutdown and low pressure core flooding for reactor cooling while maintaining reactor water level as well. Therefore, a total of three pumps delivering 954m^3/h of flow rate are provided. This flow rate satisfies the required minimum flow rate to maintain reactor water level according to the safety analysis presented in Chapter 24: Design Basis Analysis. The main configuration aspects with regard to the RHR Pump are described as follows.

- A minimum flow line is installed to protect the pumps from overheating during low flow operation. Motor-operated valves, orifices, and check valves are provided on the minimum flow pipe.
- The RHR Pumps and the discharge lines are maintained filled with pressurised water during normal operation in order to prevent time delays during initiation and water hammer from occurring.

(iii) Performance

The RHR Pump is designed to perform as follows in order to deliver the various functions of the RHR ([RHR SFC 3-1.1], [RHR SFC 3-1.2], [RHR SFC 2-1-.1], [RHR SFC 2-4.1], [RHR SFC 3-1.3], [RHR SFC 3-1.4], [RHR SFC 3-1.5] and [RHR SFC 4-7.2]).

Number:	3 units
Pump Type:	Turbo
Design Pressure:	approx. 3.43 MPa [gauge]
Design Temperature:	approx. 182°C
Rated Flow Rate:	approx. 954 m^3/h
Motor Type:	Induction Motor

The RHR Pumps are designed such that they can be initiated with the injection valves closed and reach rated flow within the time required by the safety analysis after receiving the initiation signal.

(b) RHR Heat Exchanger

(i) Purpose

The RHR Heat Exchangers are intended to remove the residual heat of the reactor by the RCW in order to deliver [RHR SFC 3-1.1], [RHR SFC 3-1.2], [RHR SFC 2-4.1], [RHR SFC 3-1.3], [RHR SFC 3-1.4] and [RHR SFC 3-1.5].

(ii) Configuration and Operation

The RHR is provided with six horizontal U-tube/shell type heat exchangers for heat removal after reactor shutdown. The tube side water is reactor coolant drawn from the RPV or S/P water. The shell side cooling water is supplied by the RCW. The heat exchangers are designed such that the water flow can be adjusted by throttle valves at the heat exchanger outlet, bypass line and cooling outlet (cooling water shell side) in order to maintain the temperature change rate below 55°C/h (operating limit during the initial stage of the reactor shutdown cooling) by changing the valve position when the heat exchanger inlet temperature is high and thus have a margin on the heat exchanged.

(iii) Performance

The RHR Heat Exchanger is designed to perform as follows in order to deliver heat removal after reactor shutdown ([RHR SFC 3-1.1], [RHR SFC 3-1.2], [RHR SFC 2-4.1], [RHR SFC 3-1.3], [RHR SFC 3-1.4] and [RHR SFC 3-1.5]).

Number: 6 units (2 per division)
Type: Horizontal U-tube Type
Design Conditions:

	Tube Side	Shell Side
Fluid:	RPV Water	Cooling Water
Flow:	approx. 954 m^3/h	approx. 1800 m^3/h
Design Pressure	approx. 3.43 MPa [gauge]	approx. 1.37 MPa [gauge]
Design Temperature	approx. 182°C	approx. 70°C
Heat Exchange Capacity:	approx. 13 MW/unit	

(3) Main Support Systems

(a) Instrumentation and Control Systems

Instrumentation

Instrumentation is provided to measure and monitor the operating conditions of the RHR components necessary for the delivery of residual heat removal after reactor shutdown. The main provisions for instrumentation are described as follows.

(i) General

RHR control devices are centralised in one area inside the MCR such that the minimum number of operators can control the system operation and the pump operating conditions, the position of valves, etc. can be easily understood.

(ii) RHR Pump

Flow-meters are mounted downstream of the RHR Heat Exchangers to monitor system flow rate and to control the minimum flow bypass valve.

(iii) RHR Heat Exchanger

Temperature indicators are provided on the inlet and outlet of the RHR Heat Exchanger in order to monitor the heat exchanger performance.

Interlocks

The main interlocks are summarised as follows.

(i) RHR Pump

An interlock is provided to the RHR Pumps to prevent them being initiated manually if the reactor water suction valves and the S/P water suction isolation valve are not fully opened.

(ii) RPV Injection Valves

An interlock is provided to prevent valves opening whenever reactor pressure is above the high pressure limits for the RHR. These valves are closed if the reactor pressure increases during operation.

(iii) Reactor Water Suction Valves

An interlock is provided to prevent reactor water suction valves from opening if the S/P water suction isolation valves, test control valves, drywell cooling spray line isolation valves and wetwell spray injection isolation valves are not fully closed to prevent draining the RPV water into the S/P.

Logic

The main logic provisions are summarised as follows.

(i) RHR Pump

The RHR Pumps start automatically after confirmation of power supply once the initiation signal has been received.

(ii) Minimum Flow Bypass Valves

The bypass valves operate automatically and simultaneously with the pumps initiation. Minimum flow bypass valves open automatically if after starting the corresponding system pumps the flow is below the minimum set point; and automatically close if the flow is above the minimum set point.

(iii) RHR Heat Exchanger Outlet Valves

The heat exchanger outlet valves receive the opening signal after the RHR has received the initiation signal.

(b) Power Supply System
  (i) Power supply for RHR components, valves, instrumentation and controllers come from the Electrical Power Distribution System.
  (ii) The RHR is connected to separated and functionally independent divisions of AC and DC power sources supplying the required power to all electrical components in each division (RHR Division A is connected to power Division I, RHR Division B is connected to power Division II and RHR Division C is connected to power Division III).
  (iii) The normal AC power supply to the RHR electrical components is provided by an independent off-site source (external grid). In addition, independent divisional power sources such as diesel generators provide a reliable alternative source of electrical power in the event of LOOP.
  (iv) The Emergency Diesel Generator of each division provides power for all RHR components in the corresponding division which require electrical supply when the normal AC power source is not available.

(c) Reactor Building Cooling Water System (RCW)
The RCW supplies cooling water to the RHR Heat Exchangers, RHR Pumps, motors, bearings and seal water cooling equipment. The RHR is connected to functionally independent and separated RCW divisions. RHR division A components are supplied cooling water by RCW division A, RHR division B components are supplied cooling water by RCW division B and RHR division C components are supplied cooling water by RCW division C.

Further details of the RCW are given in Chapter 16: Auxiliary Systems.

Assumptions, Limits and Conditions for Operation

In order to ensure that the RHR is operated within the safety limits and that the design requirements from the safety case are met during the operating regime, appropriate LCOs and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the ‘Basis of Safety Cases on Residual Heat Removal System’ (Ref 12.3-6).

- The three RHR shutdown cooling subsystems shall be operable during hot shutdown for the delivery of the SFCs claimed when required.
- The three RHR shutdown cooling subsystems shall be operable during cold shutdown for the delivery of the SFCs claimed when required (one RHR division may be inoperable after a number of hours from initial entry into hot shutdown in the process to shut down the reactor).
- One RHR shutdown cooling subsystems shall be operating and another one operable during refuelling for the delivery of the SFCs claimed when required.

The LCOs listed above are related only with the RHR function of shutdown cooling. Other LCOs for the RHR exist related to the functions of containment heat removal during faults and reactor core cooling as part of the ECCS. For these LCOs, refer to Chapter 13: Engineered Safety Features, Section 13.3.3.4 about containment heat removal systems, and Section 13.4 about ECCS.

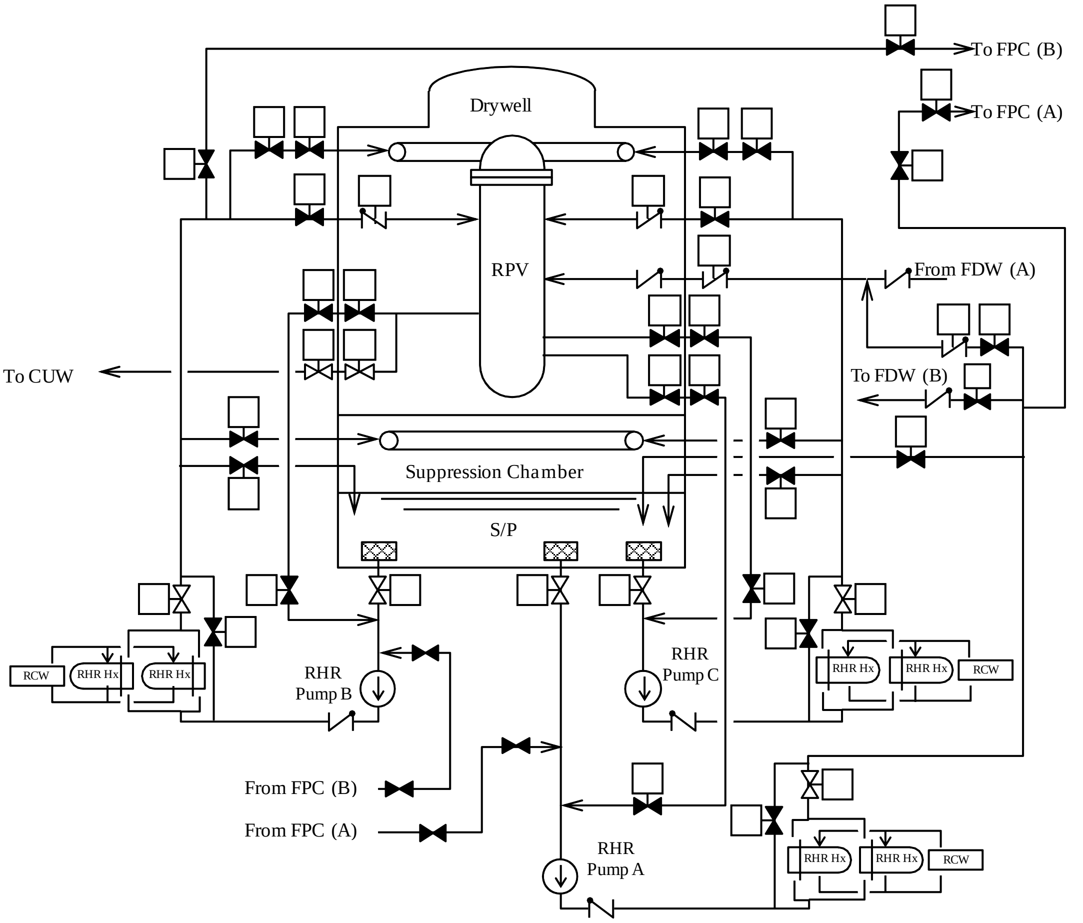

Figure 12.3-15 : Outline of the RHR

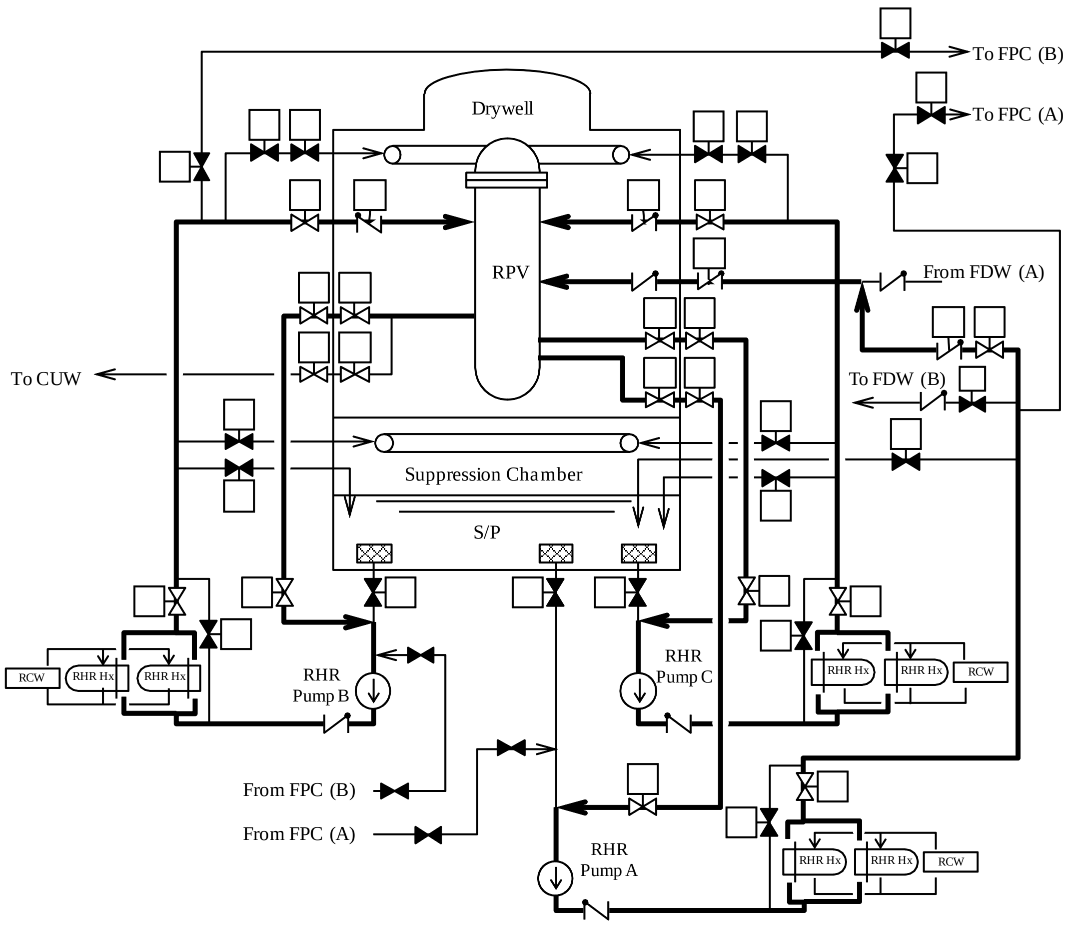

Figure 12.3-16 : Outline of Reactor Shutdown Cooling Mode Operation

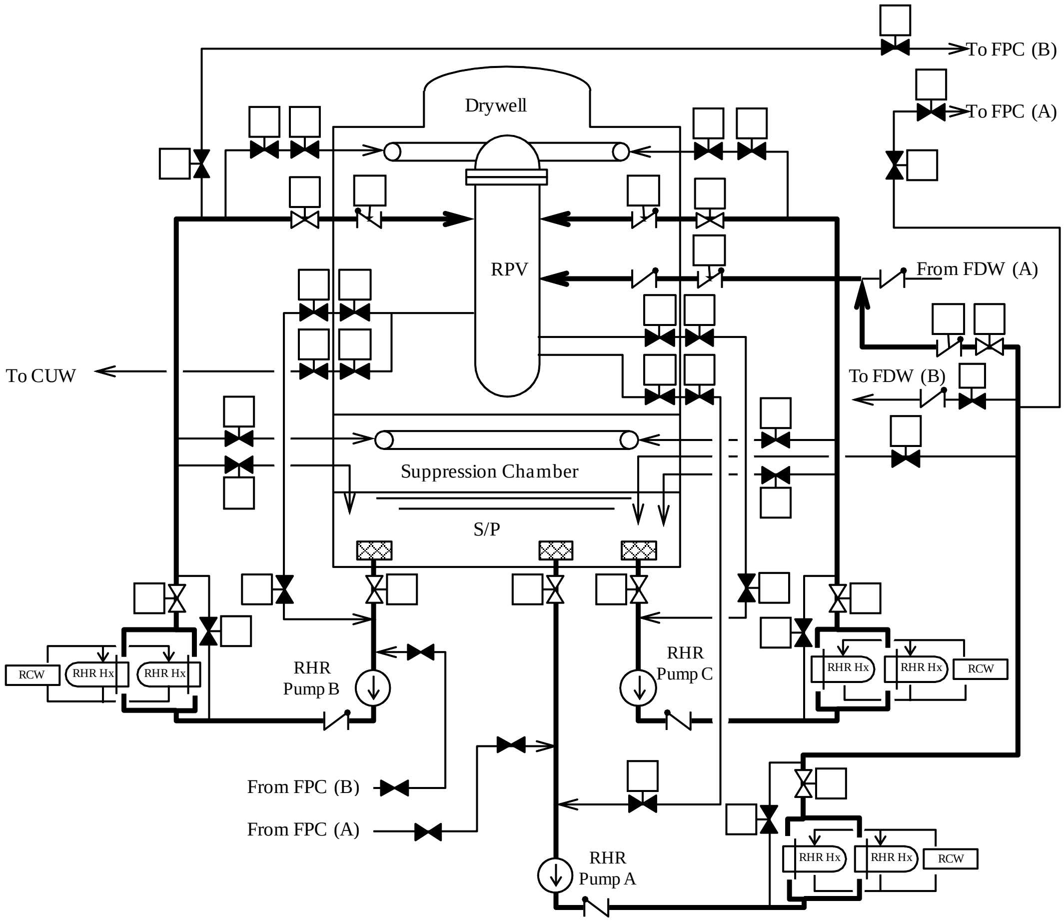

Figure 12.3-17 : Outline of LPFL Mode Operation

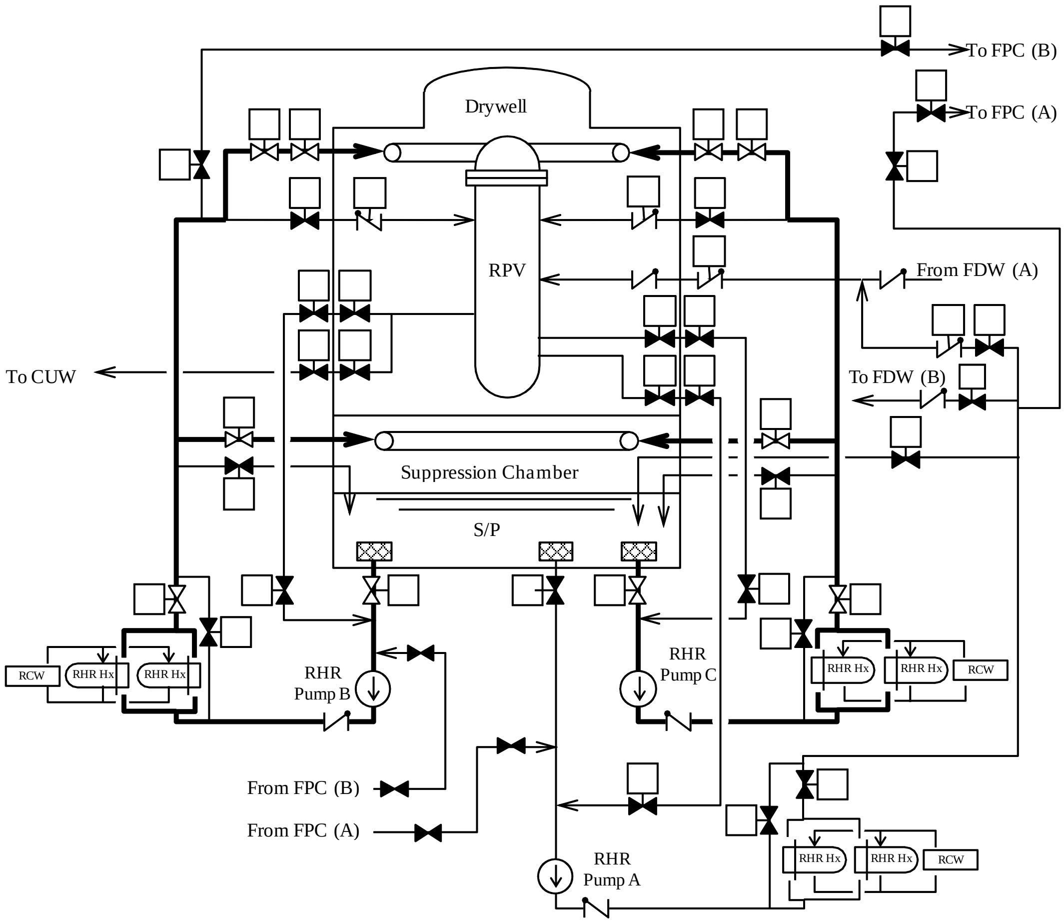

Figure 12.3-18 : Outline of PCV Spray Cooling Mode Operation

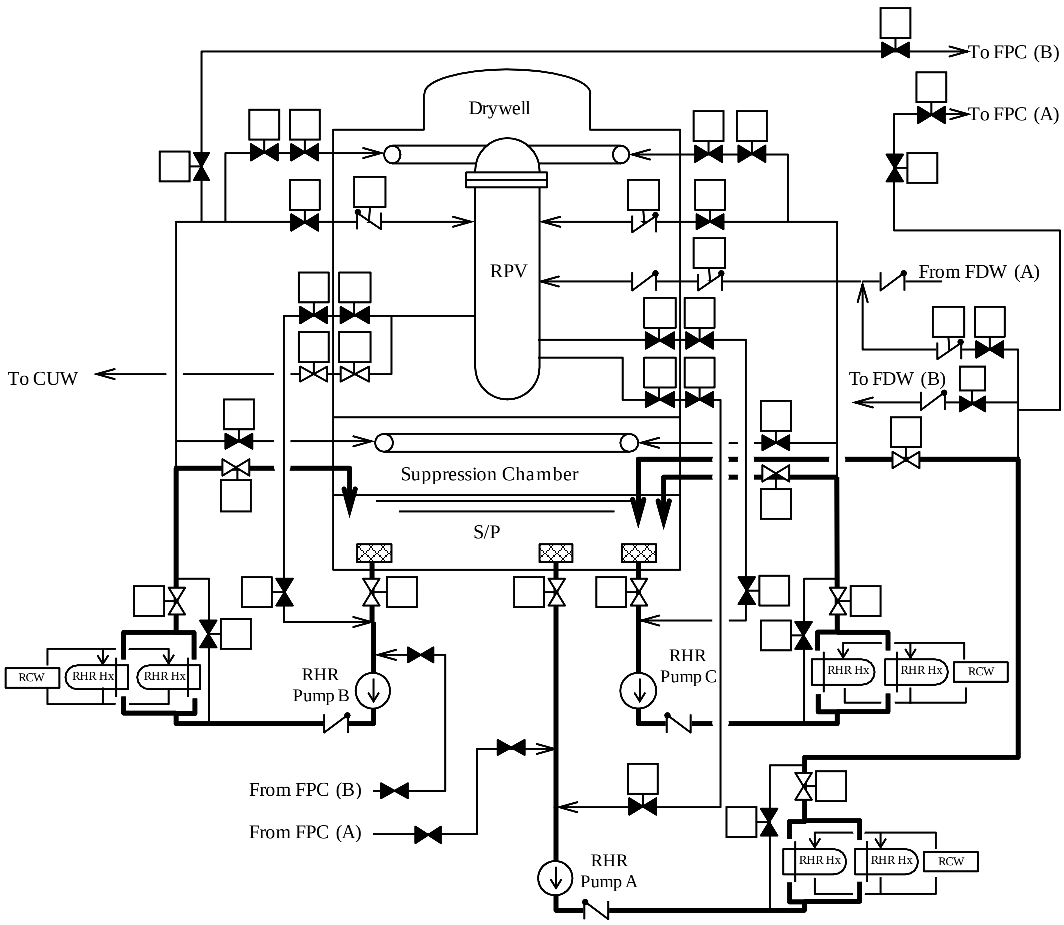

Figure 12.3-19 : Outline of Suppression Pool Cooling Mode Operation

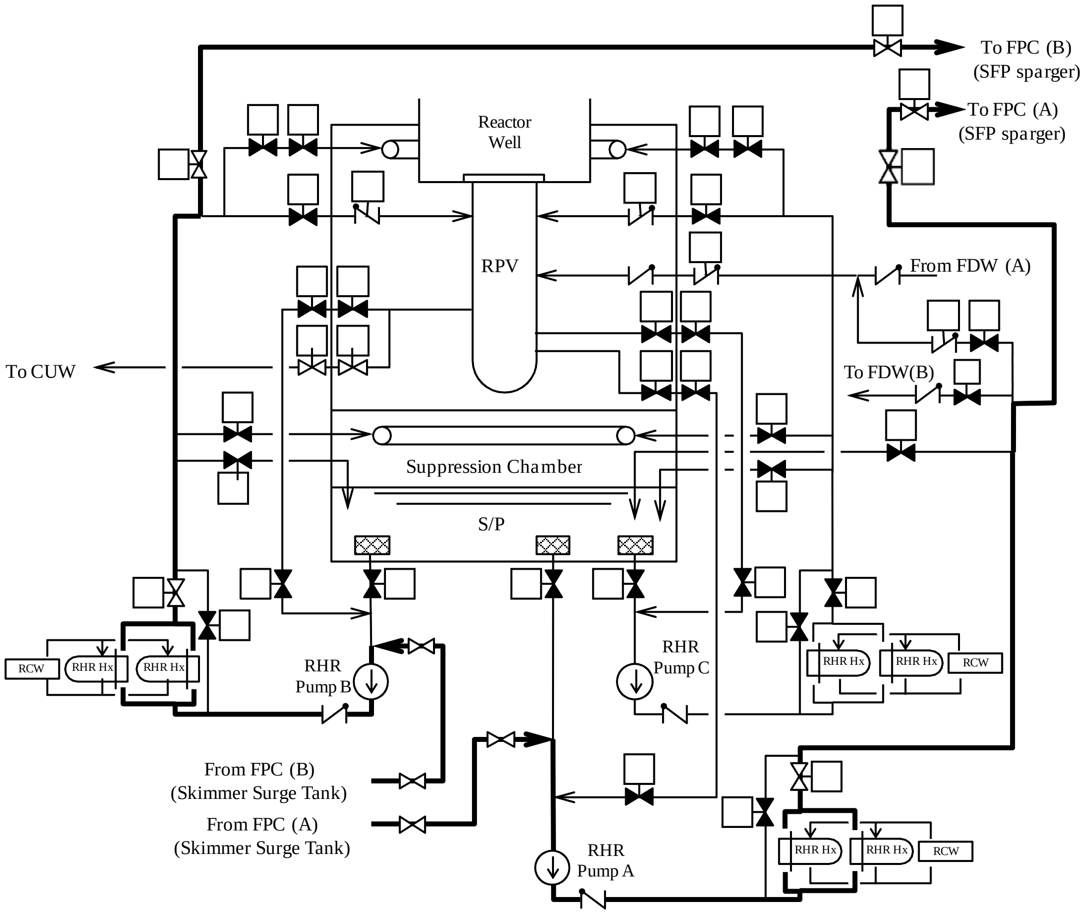

Figure 12.3-20 : Outline of Fuel Pool Cooling Mode Operation

12.3.5.5 Valves and Associated Systems

Line Valves

Line valves, such as gate, globe, and check valves, are located in the fluid systems to perform mechanical functions. Valves are components of the system pressure boundary and, having moving parts, are designed to operate reliably and to maintain the integrity of the RCPB.

The valves are designed to operate under internal pressure and temperature loading as well as external loading experienced during the various system transient operating conditions. The design criteria, the design loading, and acceptability criteria are in accordance with the ASME Code.

(1) Component Design

Line valves are manufactured standard types designed and constructed in accordance with the requirements of ASME Code Section III for Class 1, 2, and 3 valves. All materials, exclusive of replacement parts such as seals, packing, and wearing components, will endure the 60-year plant life under the environmental conditions applicable to the particular system when appropriate maintenance is periodically performed.

Actuators are sized to operate reliably under the maximum differential pressure as specified in the equipment specification.

(2) Safety Evaluation

Line valves are shop tested by the manufacturer for pressure integrity and operability. Pressure-retaining parts are subject to testing and examination requirements of ASME Code Section III. To demonstrate leak tightness against internal and external leakage past seating surfaces, maximum allowable leakage rates are stated in the equipment specifications for both back seat as well as the main seat for line valves, and tested in the shop.

Valve construction materials are compatible with the service ambient conditions, dynamic loading, and the maximum anticipated radiation dosage for the service life of the valves.

Safety related actuators are qualified for normal and accidental ambient conditions and dynamic loadings based on applicable industry standards.

(3) Inspection and Testing

Valves serving as containment isolation valves which must remain closed or open during normal plant operation are exercised periodically to assure their operability at the time of an emergency or faulted condition. Other valves, serving as a system block or throttling valves may be exercised when appropriate.

Motors used with valve actuators are furnished in accordance with applicable industry standards. Each motor actuator is assembled, factory tested, and adjusted on the valve for proper operation, position, torque switch setting, position transmitter function (where applicable), and speed requirements. Valves are tested to demonstrate adequate stem thrust (or torque) capability to open or close the valve within the specified time at specified differential pressure. Tests are used to verify no mechanical damage to valve components during full stroking of the valve. Valve suppliers must provide assurance of acceptability of equipment for the intended service based on any combination of:

- Test stand data,
- prior field performance,

- prototype testing, and
- engineering analysis.

Valve Gland Leakage Treatment System

(1) System Summary Description

This section is a general introduction to the VGL where the system roles, system functions and system configuration are briefly described. The VGL design is described in detail in the system design specifications (Ref 12.3-27) and the P&IDs (Ref 12.3-28 and Ref 12.3-29).

(a) System Roles

The purpose of the VGL is to prevent the discharge of radioactive fluids outside the RCPB by transferring the fluid leakage from the valve’s gland to the drywell LCW Sump or the S/P through the piping system.

(b) Functions Delivered

The VGL collects the fluid leakage from the valves with a gland seal (except for the valves without leakage such as bellow seal valves, etc.) and that form part of the RCPB.

(c) Basic Configuration

The VGL consists of the piping system to collect the leakage from the valves gland seal and transfer it to the LCW Sump or the S/P.

(2) Design Bases

This section describes the design bases for the VGL. The VGL has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

<u>Normal Conditions</u>

(a) The VGL collects reactor coolant leakage from the gland seal of the target valves and transfers it to the corresponding collection point during normal operation. [VGL SFC 4-12.1]
(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

<u>Fault Conditions</u>

(b) The VGL components penetrating the primary containment form a barrier to confine radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults. [VGL SFC 4-7.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified in Chapter 13: Engineered Safety Features, Section 13.3.3.2 Primary Containment Isolation System)

(3) System Design Description

This section describes the design of the VGL to support and justify the delivery of [VGL SFC 4-12.1]. Additional design description can be found in Ref 12.3-27, Ref 12.3-28 and Ref 12.3-29.

(a) Overall System Design and Operation

The VGL is designed to prevent the dispersion of reactor coolant through the gland seal of the valves forming part of the RCPB (except for the valves without leakage such as bellow seal valves, etc.). For this purpose the system is configured with a piping system to collect the gland leakage through the leak-off pipes mounted on the valves gland in order to collect it and ultimately treat it in the radwaste facilities.

The valves subjected to gland leakage collection are broadly classified as follows:

(i) The valves within the PCV including inboard MSIV
(ii) The valves within the R/B (outboard MSIV only)

A VGL piping system is configured for each division.

The piping collecting the gland leak-off from the valves within the PCV joins in one pipeline at the end to discharge into the drywell LCW Sump. On the other hand, the piping collecting the gland leak-off from valves within the R/B joins in one pipeline at the end to discharge into the S/P.
The gland leak-off from the target valves is detected through the temperature indicators provided on the leak-off collecting piping.

(b) Main Support Systems

Instrumentation and Control Systems
Temperature indicators are mounted on the leak-off pipes to record the temperature and initiate an alarm in the MCR upon high temperature.

Assumptions, Limits and Conditions for Operation

Based on the general principles for the identification of Assumptions and LCOs described in Standard Control Procedure for Identification and Registration of Assumptions, Limits and Conditions for Operation (Ref. 12.1-8 “XD-GD-0042”), limits directly related to operational aspects, ex. Operation Control, will be determined in the site specific phase and the details will commensurate with the maturity of the design. For this reason there are no applicable LCOs on this system within the scope of the GDA.

12.3.5.6 Component Supports

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

(1) Design Bases

Design loading combinations, design procedures, and acceptability criteria are as described in ASME Code section III. Flexibility calculations and seismic analysis for Class 1, 2, and 3 components are confirmed with the appropriate requirements of ASME Code Section III. Support types and materials used for fabricated support elements conform with the appropriate Sections of ASME Code Section III. Pipe support spacing guidelines in ASME Code Section III, are to be followed.

(2) Description

The use and the location of rigid-type supports, variable or constant spring-type supports, snubbers, and anchors or guides are determined by flexibility and seismic stress analyses. Component support elements are manufacturer standard items. Attachment method of the pipe to the support element will be appropriate to avoid failure of the pipe due to loading.

(3) Safety Evaluation

The flexibility and seismic analyses are performed for the design of adequate component support systems that include all loading conditions expected by each component. Spring-type supports are designed considering the additional dead weight loading due to hydrostatic testing of steam systems to prevent damage to this type of support.

(4) Inspection and Testing

After completion of the installation of a support system, all hangers and snubbers are to be visually examined to assure that they are in correct adjustment to their cold setting position. Weld inspections and standards are to be in accordance with ASME Code Section III. Welder qualifications and welding procedures are in accordance with ASME Code Section IX and the appropriate Sections of ASME Code Section III.

12.4 Reactivity Control Systems

12.4.1 Summary Description of the Reactivity Control Systems

The Reactivity Control Systems consists of the CRs and the CRD, and the SLC. This chapter summarises the design and safety functions for the CRD and SLC. The CRs are described in Chapter 11: Reactor Core.

The CRD consists of the electro-hydraulic FMCRD, the Hydraulic Control Unit (HCU), and the CRD hydraulic system. The Control Rod Drive hydraulic system is further divided into components such as the CRD Pumps, filters, piping and valves.

The SLC consists of components such as a sodium pentaborate solution storage tank, pumps, a test tank, piping and valves.

As for the reactor shutdown systems, the CRD has the function of shutting down the reactor. The reactor is shut down by inserting the CRs into the core. The CRD inserts the CRs into the core and withdraws them from the core at the speed required for normal operations to control the reactivity of core.

In an emergency, the CRD inserts the CRs into the core rapidly to scram the reactor (to bring the reactor to an emergency shutdown).

In cases where the CRs cannot be inserted, the SLC, as the back-up system of the CRD, injects a liquid neutron absorber into the reactor which inserts negative reactivity and shuts down the reactor.

12.4.2 Safety Requirements

The safety requirements for Reactivity Control Systems are described within the design bases in the respective sub-sections of each system, separately.

12.4.3 Components and Subsystems

12.4.3.1 Control Rod Drive System (CRD)

System Summary Description

This section is a general introduction to the CRD where the system roles, system functions, system configuration and modes of operation are briefly described.The CRD safety case is justified in the 'Basis of Safety Cases on Control Rod Drive System' (Ref-12.4-1). The CRD design is described in detail in the system design specifications (Ref-12.4-3) and the P&IDs (Ref-12.4-4, Ref-12.4-5 and Ref-12.4-6).

(1) System Roles
The main roles of the CRD are the following:
- (a) The CRD drives the electro-hydraulic FMCRDs through an electric motor and thereby changes the position of the CRs in the core to control the reactivity during normal operation.
- (b) During transients of the plant, the FMCRDs can be hydraulically driven by pressurised water from the HCU to rapidly insert all the CRs into the core, in an action known as a scram, and thereby shut down the reactor safely and quickly.
- (c) During normal operation the CRD Pumps supply purge water to the FMCRDs, the RIPs, and the CUW Pump while continuously maintaining the HCU Accumulators pressurised with water to ensure they are charged at high pressure for possible scrams.

(2) Functions Delivered
The CRD is designed to perform the following functions:
- (a) Each CRD, through its electric motor positions the CRs in the core depending on the control signal from the Rod Control and Information System (RC&IS) when performing normal insertion and withdrawal for control of changes in core reactivity.
- (b) The CRD through the FMCRDs implements reactor scram operation when receiving the scram signal from the RPS. The FMCRD electric motors are actuated to back up the full insertion of the CRs with the scram follow-in signal from the RPS.
  In a scram situation, the CRD opens the scram valves provided on the outlet of each HCU accumulator and thereby the pressurised water stored in the HCU accumulator is supplied to the hollow piston section of the FMCRD in the event that the scram signal was initiated. As a result, the FMCRDs are hydraulically driven and each CR is rapidly inserted into the reactor core to shut down the reactor. Once fully inserted, a latch prevents the CR from moving out of the core.
- (c) The Alternative Rod Insertion (ARI) signal, from the ATWS, opens solenoid-operated valves on the scram air header to reduce pressure in the header, allowing the HCU scram valves to open, and thus allows the CRs to be hydraulically inserted into the reactor core in case scram could not be performed upon RPS scram signal. At the same time, the CRD actuates the FMCRD electric motors to back up the full insertion of the CRs with the FMCRD run-in signal from the ATWS.
- (d) The CRD through the CRD Pumps supplies water to maintain the full pressure required in the HCU accumulators to enable a rapid scram of the reactor.
- (e) The CRD through the CRD Pumps supplies purge water to reduce radioactive contamination due to deposition of activated corrosion products contained in the reactor water inside the FMCRDs during plant normal operation.
- (f) The CRD through the CRD Pumps supplies purge water to reduce the radioactive contamination due to deposition of activated corrosion products contained in the reactor

water within the motor side of the RIPs and the CUW Pumps during plant normal operation.

(g) The CRD is utilised to pressurise the RPV when the leakage and hydrostatic test is implemented.

(h) The CRD is used to deliver the SCRRI function in conjunction with the RC&IS and the RFC, which inserts the CRs previously selected by controlling their FMCRD motor in order to mitigate power increases.

(3) Basic Configuration

The CRD consists of the following main components. Figure 12.4-1 shows an outline of the CRD configuration.

(a) FMCRD

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the CRs and hydraulic-powered rapid insertion (scram) of the CRs during abnormal operating conditions. There are a total of 205 FMCRDs mounted in housings welded into the reactor vessel bottom head.

Furthermore, the FMCRD electric motors are actuated to back up the full insertion of the CRs with the scram follow-in signal from the RPS.

(b) HCU

The hydraulic power required for scram is provided by high pressure water stored in 103 individual HCUs. Each HCU contains a nitrogen-water accumulator charged to high pressure and the necessary valves and components to simultaneously fully insert two CRs. In addition, during normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs.

(c) CRD Pumps

Through the CRD Pumps the CRD supplies clean, demineralised water which is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD Pumps also supply pressurised water for purging the RIPs and the CUW Pumps.

(d) CRD Drive Water Heater

This heater is provided to heat CRD water and maintain the scram lines connecting each HCU to its associated FMCRDs at a temperature to avoid condensation on the outside of the lines which helps avoid corrosion of the scram lines.

(e) CRD Pump Suction Filters

CRD pump suction filters are provided to prevent ingress of debris to the pumps during operation particularly following outage and commissioning.

(f) CRD Drive Water Filters

Filters are provided to ensure no corrosion products or other contamination is present in the water supplied to the FMCRDs, RIPs and CUW to protect the components downstream the CRD Pump.

(g) CRD Charging Header Accumulator

The charging header accumulator is provided to ensure water pressure is maintained in the event of a CRD Pump trip, while the standby pump starts to avoid any spurious reactor scrams due to low charging water pressure.

(h) HCU Nitrogen Gas Charging Equipment

This is required to maintain the nitrogen overpressure in the HCUs.

(i) Valves, piping, instrumentation, and controllers.

(4) Modes of Operation

(a) Normal Operation Mode

Normal operation of the CRD is defined as those periods of time when no control rod drives are in motion. Under this condition, the CRD provides charging pressure to the HCUs and supplies purge water to the CR drives, RIPs and CUW Pumps.

(b) Control Rod Insertion and Withdrawal Mode

The CRD receives the control signal from the RC&IS and actuates the FMCRD electric motors to drive the CRs according to the specified insertion/withdrawal steps when implementing normal insertion/withdrawal of the CRs for control of changes in core reactivity and normal reactor start-up and shutdown.

(c) Scram Drive Mode

Upon loss of electric power to both scram pilot valve solenoids, the scram valve in the associated HCU opens to apply the hydraulic insert forces to its respective FMCRDs using high pressure water stored within the pre-charged accumulator.

The water is driven by the pressurised nitrogen in the accumulator and nitrogen bottle. The CRs are driven fully into the reactor core inserting adequate negative reactivity to shutdown the reactor.

(d) Scram Completion Mode

The RC&IS transmits the control signal to the FMCRDs after receiving the control signal from the RPS. In consequence the FMCRDs actuate the electric motors in order to initiate the scram follow-in action.

(e) ARI

The ARI function of the CRD System provides an alternate means for actuating hydraulic scram that is diverse from the RPS. The signals to initiate the ARI are high reactor dome pressure or low reactor vessel water Level 2 or manual operator action. Following receipt of any of these signals, solenoid-operated valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open. The FMCRDs then insert the CRs hydraulically in the same manner as the RPS initiated scram. The same signals that initiate ARI will simultaneously actuate the FMCRD motors to insert the CRs electrically.

Design Bases

This section describes the design bases for the CRD. The CRD has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

Normal Conditions

(1) The CRD through its FMCRD is part of the principal means (in conjunction with the RC&IS) to deliver control rod insertion/withdrawal in plant normal operation conditions, the loss of which could lead to the demand of a Category A safety function (reactor scram).[CRD SFC 1-7.1]
(This normal operation function is categorised as Safety Category B and the components to deliver it are designed to meet Safety 3 requirements)

(2) The CRD through its FMCRD is the principal means to deliver maintenance of core sub-criticality in plant normal operation conditions.[CRD SFC 1-4.1]
(This normal operation function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(3) The CRD is the principal means to prevent excessive reactivity insertion caused by a CR drop event when the CR is separated from the ball nut. [CRD SFC 1-1.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(4) The CRD is capable of supplying makeup coolant to the reactor to compensate for a small leakage and prevent if from resulting in a LOCA. [CRD SFC 2-3.1]
(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

(5) The CRD through its CRD Hydraulic System supports the delivery of reactor rapid shutdown through CRs hydraulic insertion by providing charging water to the HCUs. [CRD SFC 5-20.1]
(This function is necessary for performing emergency shutdown of the reactor, however in the event of loss of this function, reactor shutdown is automatically performed. Therefore it is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

(6) Full insertion of the FMCRD with the electric motor after reactor was shut down by hydraulic insertion of the control rods provides recovery to unlatched state of the CRs. [CRD SFC 5-20.2]
(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

(7) The CRD piping and components contain radioactive material. A breach could lead to a release of radioactive material of dose consequences that are relatively low. [CRD SFC 4-4.1]
(This function is categorised as Category B and the components to deliver it are designed to meet Class 3 requirements)

Normal and Fault Conditions

(8) The CRD portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [CRD SFC 4-1.1]

(This function is categorised as Category A and the components to deliver it are designed to meet Class 1 requirements. It is developed and justified as explained in section 12.3.3.1)

Fault Conditions

(9) The CRD is the principal means to provide reactor rapid shutdown under RPS signal in conjunction with the CRs by performing CRs insertion (actuation known as Scram), so that fuel design margins are not exceeded in the event of frequent faults and infrequent faults requiring reactor shutdown. [CRD SFC 1-3.1]
(This mitigation function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(10) The CRD portions operated under the ARI signal are part of the secondary means to provide alternative reactor shutdown in the event of a frequent fault where reactor shutdown by Scram has failed (event known as Anticipated Transient Without Scram – ATWS). [CRD SFC 1-5.1]
(This mitigation function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements)

(11) The FMCRD Run-In mode of the FMCRD provides an alternative means of reactor shutdown by inserting the CRs with the FMCRD in the event reactor shutdown could not be achieved by RPS Scram, ARI and SLC. [CRD SFC 1-5.2]
(This mitigation function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

(12) The CRD in conjunction with the RC&IS is part of the means to deliver the SCRRI function which is an alternative means of controlling reactivity by inserting the CRs previously selected by controlling their FMCRD motor. [CRD SFC 1-5.3]
(This function is categorised as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements)

(13) The CRD is the principal means to maintain the control rods inserted when shutdown by Scram in order to maintain the core sub-criticality. [CRD SFC 1-4.2]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(14) The CRD through its FMCRD is the principal means to prevent excessive reactivity insertion by prevention of control rod ejection. [CRD SFC 1-1.2]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(15) The CRD through its FMCRD is the principal means to prevent excessive reactivity insertion by limiting CRs drop speed. [CRD SFC 1-1.3]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements)

(16) The CRD components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults. [CRD SFC 4-7.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. It is developed in Chapter 13: Engineered Safety Features, Section 13.3.3.2 Primary Containment Isolation System)

System Design Description

This section describes the design of the CRD to support and justify the delivery of [CRD SFC 1-7.1], [CRD SFC 1-1.1], [CRD SFC 1-4.1], [CRD SFC 2-3.1], [CRD SFC 4-1.1], [CRD SFC 4-4.1], [CRD SFC 1-8.1], [CRD SFC 1-8.2], [CRD SFC 1-3.1], [CRD SFC 1-3.2], [CRD SFC 1-3.3], [CRD SFC 1-5.1], [CRD SFC 1-5.2], [CRD SFC 1-5.3], [CRD SFC 1-5.4], [CRD SFC 1-4.2], [CRD SFC 1-1.2], [CRD SFC 1-1.3], and [CRD SFC 4-7.1]. Additional design description can be found in Ref-12.4-3, Ref-12.4-4, Ref-12.4-5 and Ref-12.4-6.

(1) Overall System Design and Operation

(a) Normal Operation Mode

Normal operation of the CRD is defined as those periods of time when no control rod drives are in motion. Under this condition, the CRD provides charging pressure to the HCUs to maintain them at high pressure standby conditions so they are ready for scram if required.

Two CRD Pumps installed in parallel (normally one in operation and the other on standby) supply the system with water from the CFDW and/or the Condensate Storage Tank (CST) depending on the operational conditions of the reactor.

Water is provided by the condenser spill over water from the CFDW during reactor normal operation.

Water is supplied from the CST during reactor shutdown or start-up.

The system water is processed by redundant filters in both the pump suction and discharge lines.

In order to maintain the ability to scram, the charging water line maintains the accumulators at high pressure. The scram valves remain closed except during and after scram, so during normal operation no flow passes through the charging water header. Pressure in the charging water header is monitored continuously. A significant degradation in the charging header pressure causes a low pressure warning alarm and rod withdrawal block by the RC&IS. If further degradation in pressure occurs, the RPS causes a reactor scram.

Pressure in the pump discharge header downstream of the CRD water filters is also monitored continuously. Low pressure in this line is used to indicate that the operating pump has failed or tripped. If it should occur, automatic start-up of the standby pump is initiated and the system is quickly repressurised. This prevents the malfunctioning of the operating pump from causing a spurious reactor scram on low charging water header pressure, an event which would otherwise be a direct consequence of the malfunction.

The CRD Charging Header Accumulator and the check valves installed in the charge water line are capable of maintaining the line pressurised until the standby CRD Pump starts up and provides sufficient pressure.

(b) Scram Drive Mode

Upon loss of electric power to both scram pilot valve solenoids, the scram inlet valve in the associated HCU opens to apply the hydraulic insert forces to its respective FMCRDs using high pressure water stored within the previously charged HCU Accumulator (the nitrogen-water accumulator previously having been pressurised with charging water from the CRD Pumps). Once the hydraulic force is applied, the hollow piston within the

FMCRD disengages from the ball-nut and inserts the CR rapidly to deliver reactor shutdown and maintenance of core sub-criticality. The water displaced from the drive is discharged into the RPV. Indication that the scram has been successfully completed (all rods full-in position) is displayed to the operator by a signal from the RC&IS.

(c) Alternative Scram Mode
In addition to the scram pilot valves, two alternative means of forcing the scram inlet valve to open are provided. These are the Backup Scram Pilot Valves and the ARI Electromagnetic valves.
The Backup Scram Pilot Valves are two three-directional electromagnetic valves installed in series with the scram pilot valve air line to assure the insertion of the CRs even if pressurized air is not discharged from the scram valve actuator due to any failure of the scram pilot valve when scram is initiated.

In addition to the scram pilot valves and the backup scram pilot valves which operate by an RPS signal, the ARI Electromagnetic Valves operated by an ARI signal from the ATWS are provided as well on the scram pilot valve air line. The ARI valves are aimed to discharge the pressurized air from the scram pilot valve air header and the scram valve actuator.

(2) Equipment Design and Operation
(a) FMCRD
(i) Purpose
The FMCRD is intended to deliver safety functions [CRD SFC 1-7.1], [CRD SFC 1-1.1], [CRD SFC 1-4.1], [CRD SFC 1-8.1], [CRD SFC 1-8.2], [CRD SFC 1-3.1], [CRD SFC 1-3.3], [CRD SFC 1-5.1], [CRD SFC 1-5.3], [CRD SFC 1-5.4], [CRD SFC 1-4.2], [CRD SFC 1-1.2], [CRD SFC 1-1.3], [CRD SFC 2-3.1] and [CRD SFC 4-1.1]. Representative functions provide control rod insertion and withdrawal during normal operation and scram in the event of design basis faults by rapid insertion.

(ii) Configuration and Operation
There are a total of 205 FMCRDs, one for each CR. The FMCRD penetrates the bottom head of the RPV. The FMCRD consists of the CRD housing welded into the bottom head of the RPV and internal components.

The FMCRD used for positioning the CR in the reactor core is a mechanical/hydraulic actuated mechanism. The CR blade is attached to a hollow piston which rests on top of a ball-screw and ball nut assembly within the CRD housing. The electric motor-driven ball screw and ball-nut are capable of positioning the CR during normal operation according to the signals from the RC&IS. In addition, hydraulic pressure is used for scrams after receiving the scram signal from the RPS or the ARI signal.

During fault conditions, each single HCU powers the scram action of two associated FMCRDs. Upon scram valve initiation, high pressure nitrogen from the HCU raises the piston within the accumulator, forcing water through the scram piping. This water is directed to each FMCRD connected to the HCU. Inside each FMCRD, high-pressure water lifts the hollow piston off the ball-nut and drives the CR into the core. Departure from the ball-nut automatically releases spring-loaded latches in the hollow piston that engage slots in the guide tube. These latches are redundant and support the CR in the inserted position to prevent excessive reactivity insertion due to an eventual drop of the CR. The CR cannot be withdrawn until the ball-nut is driven up and engaged with the hollow piston.

A bayonet coupling is located between the CR and the FMCRD to engage them. Once locked, the drive and rod form an integral unit that can only be unlocked manually during an outage by specific procedures before the components can be separated.

(iii) Performance
The FMCRD components for scram are designed to be hydraulically actuated by the HCU and thus fully insert the CRs within the times specified as follows to deliver reactor rapid shutdown and maintenance of core sub-criticality.

Table 12.4-1 : FMCRD Performance

Percent of CR insertion	Time (s)
60%	≤ 1.44
100% (full insertion)	≤ 2.80

(b) HCU
(i) Purpose
The HCU is a part of the CRD hydraulic system and the purpose is the following:

- Supplying purge water to the FMCRD mechanism for preserving the function of electrical positioning and insertion of the CRs.
- Supplying high pressure water at a high speed to rapidly insert the CRs (scram) under an emergency. This is related to [CRD SFC 1-3.1] and [CRD SFC 1-5.1].

(ii) Configuration and Operation
Each HCU furnishes pressurised water for scram, on signal from the RPS, to the two associated FMCRDs. There are 103 HCUs in total, of which 102 units actuate two FMCRDs and one unit actuates one FMCRD. Additionally, each HCU provides the capability to adjust purge flow to the two associated FMCRDs.

The HCU basically consists of a purge water solenoid valve, a scram pilot valve, a scram valve, the accumulator and the nitrogen gas bottle. Each HCU (except for the one driving only one FMCRD) is capable of storing the energy required to force the scram of two FMCRDs.

The scram pilot valve is operated by the signal from the RPS. The scram pilot valve consists of two three-way solenoid valves to control the scram valve. The scram pilot valve is solenoid-operated and is normally energised. Upon loss of electrical signal to the solenoids (loss of power supply), the inlet port closes and the exhaust port opens to assure fail-safe condition. The scram pilot valve is designed so that both solenoids must be de-energised before air pressure can be discharged from the scram valve actuator. This prevents the inadvertent scram of both drives associated with a given HCU in the event of a failure of one of the pilot valve solenoids.

The scram valves are provided in order to assure a reliable scram when required. The scram valve opens to supply pressurised water to the bottom of the hollow piston. This valve is operated by an internal spring and air pressure. The scram valves are kept closed by the effect of the air pressure during normal operation. The scram valves are designed such that in the event of loss of electric power to the scram pilot valve or loss of air supply, the actuator discharges the pressurised air and the valves open to perform scram (fail-safe design). The scram valves are opened by the pressurised air discharge from the actuator upon either both of the scram pilot valve solenoids being de-energised, or the scram pilot valve air line being depressurised by the backup scram pilot valves or the ARI electromagnetic valves.

The scram accumulator stores sufficient energy to fully insert two CRs. The accumulator is a cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging header prevents loss of water pressure in the event that supply pressure is lost. In order to ensure that the accumulator is always available to perform scram, instrumentation is installed in the HCU to confirm the nitrogen gas is maintained at high pressure and there is no water leakage. Isolation is possible at each HCU and a ball check valve is installed on each FMCRD to ensure leakage is minimised in the event of a scram pipe-break.

(iii) Performance
The HCU accumulators are designed as follows in order to deliver the performance required for scram.

Table 12.4-2 : HCU Accumulator Parameters

Accumulator capacity	approx. 66 L (water side)
Nitrogen Gas Bottle capacity	approx. 200 L
Charging Water Pressure	approx. 15 MPa [gauge]

The capacity of the water side and nitrogen side of the HCU accumulators is sufficient to insert the two associated CRs of each HCU within the required time.

(c) CRD Pump
(i) Purpose
The CRD Pump supplies purge water to reduce the levels of radioactive contamination due to leakage of reactor water into the motor side of the RIPs and the CUW Pumps during normal operation.

The CRD Pump supplies purge water to the FMCRDs and thus prevent the deposition of crud contained in the reactor water inside the FMCRDs during normal operation.

The CRD Pump supplies charging water to the HCU during normal operation to maintain them charged at the required pressure so they are ready for reactor shutdown by scram action if required. This is related to [CRD SFC 1-3.2] and [CRD SFC 1-5.2].

(ii) Configuration and Operation
One supply pump pressurises the CRD with water from the CFDW or the CST while the spare pump is on standby.

(iii) Performance
The pumps are designed as indicated below such that they can supply the rated flow and head required for pressurising the HCU Accumulators (approx. 15MPa [gauge]) during normal operation and after scram completion. The pumps are operated continuously at the rated flow and pressure for purging during normal plant operation.

Table 12.4-3 : CRD Pump Parameters

Capacity for HCU Accumulator Charge	approx. 46 m^3/h
Head for HCU Accumulator Charge	Head: ≥1,420 m (approx.)

(3) Main Support Systems

(a) Instrumentation and Control Systems

(i) Instrumentation

Instrumentation is provided to measure and monitor the operating conditions of the CRD components necessary for the delivery of the safety functions. The status, measurements and alarms of the components and valves to be remotely operated are generally displayed in the MCR. The main provisions for instrumentation and control are described as follows:

- Charging water header inlet pressure (start-up of standby CRD Pump if low),
- charging water header pressure (CR withdrawal block if low, reactor scram if low-low),
- scram pilot valve air header inlet pressure, and
- HCU accumulator pressure.

(ii) Control

- The HCU scram pilot valve is actuated (for discharge) by the scram signal from the RPS and thereby the scram valve is opened to implement scram. For further details see Chapter 14: Control and Instrumentation.
- The backup scram valves at the scram pilot valve air line are actuated (discharge) by the scram signal from the RPS and thereby the scram valve opened to implement scram. For further details see Chapter 14: Control and Instrumentation.
- The ARI electromagnetic valves at the scram pilot valve air line are actuated (discharge) by the ARI signal from the HWBS and thereby the scram valve opened to implement hydraulic insertion of the CRs. For further details see Chapter 14: Control and Instrumentation.
- The CRD Pump on standby is automatically started up if the charging water header inlet pressure decreases in order to maintain the line pressure at an equal or higher level than the scram trip set value to prevent spurious scram.
- The charging water header low-low pressure signal is transmitted to the RPS for implementing the reactor scram before the scram function is lost.

(b) Power Supply System

(i) Scram Function Delivery

The design of the CRD is fail-safe, and therefore power supply is not required for the delivery of shutdown by scram ([CRD SFC 1-3.1], [CRD SFC 1-5.1]).

(ii) Other Functions Delivery

Power supply for CRD components, valves, instrumentation and controllers come from the Electrical Power Distribution System. The power supply to the CRD Pumps and FMCRDs is summarised as follows.

- The normal AC power supply to the CRD electrical components is provided by an independent off-site source (external grid). In addition, the CRD is connected to the emergency power supply system to supply AC and DC power to CRD components, valves, instruments and controllers in the event of LOOP.

- The CRD Pump electric motors are connected to two independent divisions of the emergency power supply system to ensure continuous operation even in the event of LOOP.
- The electric motors for the FMCRD are connected to the three independent divisions of emergency power supply (Divisions I, II and III).

Assumptions, Limits and Conditions for Operation

In order to ensure that the CRD is operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCOs and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the 'Basis of Safety Cases on Control Rod Drive System' (Ref-12.4-1).

- No CR stuck and minimum number of CRs shall be operable during start-up and power operation for the delivery of the SFCs claimed when required.
- CR average and individual scram times shall be within the prescribed limits during start-up and power operation for the delivery of the SFCs claimed when required.
- Each CR accumulator shall be operable during start-up and power operation for the delivery of the SFCs claimed when required.

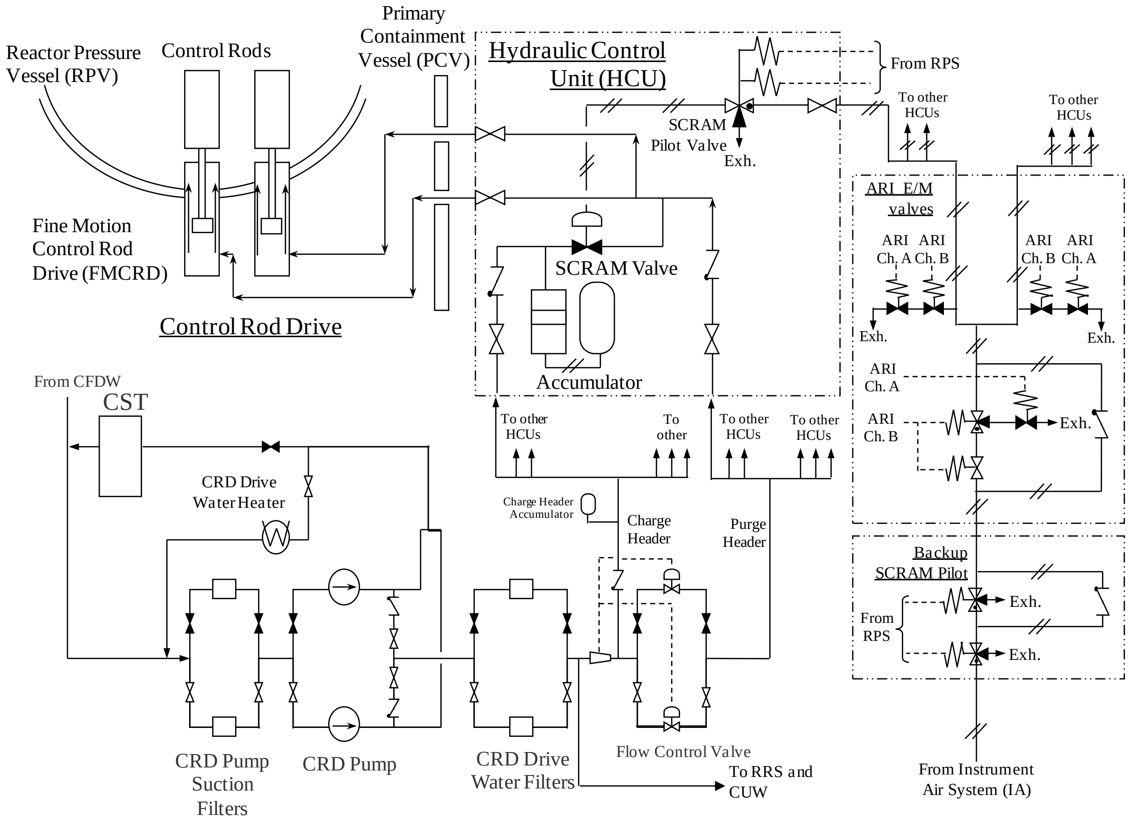

Figure 12.4-1 : Outline of the CRD

12.4.3.2 Standby Liquid Control System (SLC)

System Summary Description

This section is a general introduction to the SLC where the system roles, system functions, system configuration and modes of operation are briefly described. The SLC safety case is justified in the 'Basis of Safety Cases on Standby Liquid System' (Ref-12.4-2). The SLC design is described in detail in the system design specifications (Ref-12.4-7) and the P&IDs (Ref-12.4-8).

(1) System Roles
The SLC is the secondary means to provide reactor shutdown and maintain sub-criticality if the reactor cannot be shut down by scram action in an event known as ATWS.

(2) Functions Delivered
The main role of the SLC is to inject a neutron absorbing solution to provide sufficient negative reactivity into the core to shut down the reactor in a safe manner from full power operation to cold shutdown conditions by automatic initiation in the unlikely event that CRs insertion is not available. Sodium pentaborate solution is used as a neutron absorber.

(3) Basic Configuration
(a) The SLC injects the neutron absorber into the core from the SLC Storage Tank through one of the HPCF flooder spargers.
The SLC consists of the following components:

(i)	SLC Storage Tank	1 unit
(ii)	SLC Test Tank	1 unit
(iii)	SLC Pump	2 units
(iv)	Motor-operated injection valve	2 units
(v)	Piping and Valves	1 set
(vi)	Instruments and Control Components	1 set

Figure 12.4-2 shows an outline of the SLC.

(4) Modes of Operation
The SLC performs the following operation modes:
(a) <u>Standby Mode</u>
During normal plant operation, the SLC lines are in a standby condition with the motor-operated valves in their normally open or normally closed positions. In this mode, the lines from the SLC Storage Tank outlet valves to the motor-operated injection valves are filled with water from the Makeup Water Purified System (MUWP).

(b) <u>Reactor Injection mode</u>
This mode is automatically actuated by the ATWS signals from the HWBS in the case that the reactor cannot be shut down by the CRs. The sodium pentaborate solution is injected from the SLC Storage Tank through the SLC Pump, the injection motor operated valve, and the HPCF sparger into the core.

Design Bases

This section describes the design bases for the SLC. The SLC has been designed to meet the following SFCs. The relation between the SFCs put on this system and the high level claims is shown on Appendix A.

Normal and Fault Conditions

(1) The SLC portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL. [SLC SFC 4-1.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. This safety function is developed and justified as explained in section 12.3.3.1)

Fault Conditions

(2) The SLC is the secondary means to provide reactor shutdown without CRs insertion, from full power operation to cold sub-critical condition by injecting the neutron absorbing solution into the reactor core in the event of ATWS design basis fault. [SLC SFC 1-5.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements)

(3) The SLC is the secondary means to maintain the reactor subcriticality without CRs insertion by injecting the neutron absorbing solution into the reactor core in the event of ATWS design basis fault. [SLC SFC 1-4.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements)

(4) The SLC components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults. [SLC SFC 4-7.1]
(This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 1 requirements. It is developed in Chapter 13: Engineered Safety Features, Section 13.3.3.2 Primary Containment Isolation System)

System Design Description

This section describes the design of the SLC to support and justify the delivery of [SLC SFC 1-5.1] and [SLC SFC 1-4.1]. Additional design description can be found in Ref-12.4-7 and Ref-12.4-8.

(1) Overall System Design and Operation

The SLC is a redundant, independent reactor shutdown system to operate as a back-up of the CRD.

The SLC injects the neutron absorber into the core from the SLC Storage Tank.

Two trains of the dynamic equipment (pumps, motor-operated injection valves) are provided to assure sufficient redundancy. The system is designed such that the specific functions can be implemented with either of the trains operating.

(2) Equipment Design and Operation

(a) SLC Storage Tank

(i) Purpose

The purpose of the SLC storage tank is to store the sodium pentaborate solution to be injected into the RPV in order to deliver [SLC SFC 1-5.1] and [SLC SFC 1-4.1].

(ii) Configuration and Operation

The SLC Storage Tank is a vertical cylindrical type of tank provided with a hatch and suitable access arrangements to load the sodium pentaborate solution.

Auxiliary equipment including electric heaters, air spargers, and various nozzles are mounted.

Two electric heaters are installed inside the Storage Tank. Each heater is designed to be capable of independently maintaining an adequate tank temperature to prevent any precipitation of sodium pentaborate solution.

Two tank solution outlet connections are mounted on the tank side surface to assure that the outlet will not be plugged by any foreign material that may inadvertently be added into the tank.

Stainless steel lining is applied to parts which have interfaces with liquids.

(iii) Performance

The SLC Storage Tank is designed to perform as follows in order to ensure the delivery of [SLC SFC 1-5.1] and [SLC SFC 1-4.1].

The capacity of the SLC Storage Tank is set up based on the boron concentration necessary to maintain the sub-critical condition of the reactor plus margins. The storage capacity of the SLC is determined as the necessary quantity of the sodium pentaborate solution, which contains the boron quantity required to achieve subcriticality.

- Number: 1 unit
- Type: Vertical cylindrical type
- Capacity: 28.7 m^3

(b) SLC Pump
(i) Purpose
The purpose of the SLC pump is to supply the boron solution from the SLC tank to the RPV in order to deliver [SLC SFC 1-5.1] and [SLC SFC 1-4.1].

(ii) Configuration and Operation
The SLC is provided with two 100% flow capacity injection pumps in parallel. The pumps are triplex plunge type.

The SLC Pumps start with a certain delay after lubricant pressure reaches the established set point.

The suction and discharge connections of the pumps are flange connection type.

(iii) Performance
The SLC Pump is designed to perform as follows in order to ensure the delivery of [SLC SFC 1-5.1] and [SLC SFC 1-4.1].

The flow rate of the SLC Pump is set up such that the boron concentration variation rate in the reactor water satisfies the minimum negative reactivity insertion rate of the reactor plus margins. The SLC Pump is designed to inject the sodium pentaborate solution in SLC Storage Tank within the time necessary to satisfy the required boron concentration variation rate in the reactor water mentioned before.

Each pump is capable of pumping the rated injection flow into the reactor at all reactor operating pressures ranging from 0MPa [gauge] to the maximum pressure required for this system.

- Number: 2 units
- Pump Type: Reciprocating
- Flow Rate: 11.4 m^3/h

(3) Main Support Systems
(a) Instrumentation and Control Systems
The SLC is controlled by the Class 2 HWBS.The main instrumentation and control provisions related to SLC operation from the performance and reliability points of view are summarised as follows.
(i) Instrumentation
- All important equipment and valve conditions, measured parameters and alarms are indicated in the MCR.
- The manual valves (test valve, test tank outlet valve) position during functional test mode is indicated in the MCR, in order to be capable of confirming the open/closed state of the valves.
- The monitored items are shown in the table below.

Table 12.4-4 : Monitored Items of the SLC

No.	Item	Indicator	Alarm	Control
1	Water Level of SLC Storage Tank	MCR Local	High/Low	-
2	Temperature in SLC Storage Tank	Local	High/Low	Heater on/off control
3	Discharge Pressure of SLC Pump	MCR Local	-	

(ii) Control

The main control provisions related to the delivery of the SLC safety functions are summarised as follows.

- The SLC Pumps are initiated and the SLC sodium pentaborate solution injection valves and Storage Tank outlet valves are opened automatically upon receipt of high reactor pressure or low reactor water level in the event that reactor shutdown by CR insertion is unavailable.
- The CUW is isolated from the NB by closure of isolation valves after ATWS condition is detected to avoid unnecessary purification of boron solution. After that, the SLC pump is initiated.
- The SLC Pump can be shut off and the SLC Storage Tank outlet valves and injection valves can be closed manually by the operator in the event all control rods insertion has been confirmed by turning the manual switch to stop position.
- The SLC Pumps, SLC Storage Tank outlet valves and the SLC injection valves are designed such that they can be operated for functional testing. The SLC Pumps discharge pressure is indicated in the local panel to confirm their operating conditions. Moreover, the SLC injection valves and the SLC Storage Tank outlet valves are interlocked so as not to open simultaneously when carrying out local opening/closing tests to prevent an inadvertent injection of sodium pentaborate solution into the reactor.
- Two heaters provided inside the SLC Storage Tank prevent precipitation of sodium pentaborate in the tank and control automatically the solution temperature.

(b) Power Supply System

The SLC is designed such that it can operate even upon LOOP. The power supplies required for the pumps, solution injection valves, storage tank outlet valves, electric heaters and control and instrumentation are provided by the B/B Class 2 electrical power supply system (Back-up Building Generator).

(c) MUWP

Water filling equipment is provided for filling the system between the SLC Storage Tank outlet valve and the injection valve with water from the MUWP, and maintain it filled at all times during system standby mode. It minimises the time delay to fill the pump discharge line with water, prevents water hammer at the system start-up, and prevents pentaborate solution leakage from the SLC Storage Tank outlet valve into the system.

Assumptions, Limits and Conditions for Operation

In order to ensure that the SLC is operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCOs and surveillance requirements to ensure the LCOs are met as well as corrective actions (measures) to follow when the LCOs are not met are defined. This information is described in detail in the corresponding section of the ‘Basis of Safety Cases on Standby Liquid Control System’ (Ref-12.4-2).

- Two SLC subsystems shall be operable during start-up and power operation for the delivery of the SFCs claimed when required.

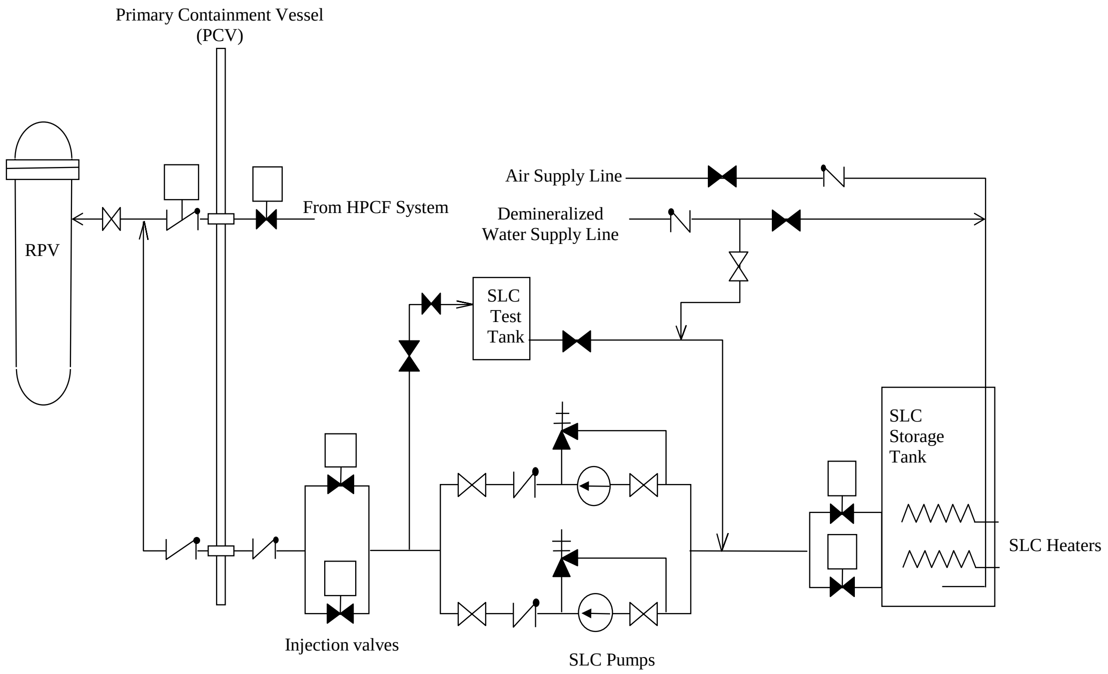

Figure 12.4-2 : Outline of the SLC

12.5 Summary of ALARP Justification

This section presents a high level overview of how the ALARP principle has been applied for the RCSs, the Reactivity Control Systems and the Associated Systems covered by Chapter 12, and how this contributes to the overall ALARP argument for the UK ABWR.
Chapter 28: ALARP Evaluation presents an overview of how the UK ABWR design has evolved, and how this evolution contributes to the overall ALARP case. The approach to ALARP during GDA is further described in the GDA ALARP Methodology (Ref 12.5-1) and the SCDM (Ref 12.1-5).
For the mechanical systems which make up this chapter, this ALARP methodology has been embedded within the design process (Ref 12.5-2). This places requirements on designers to consider ALARP through a comprehensive check which includes such elements as the identification of Relevant Good Practice (RGP) and Operational Experience (OPEX), gap analysis, risk assessment, optioneering and design review throughout the design process.

The most significant nuclear safety risks associated specifically with the RCSs, the Reactivity Control Systems and the Associated Systems of the UK ABWR are:

- Breach of the RCPB allowing leakage of radioactive reactor coolant into either the primary or secondary containment, e.g. due to a pipe break.
- Inability to provide sufficient core cooling in normal conditions, due to low core flow rate or a significant loss of core coolant inventory.
- Inability to adequately control the overall core reactivity or the local core power distributions.
- Inability to provide rapid shutdown of the core (Scram) using control rods in fault conditions, or to maintain the core sub-critical following normal shutdown or Scram.
- Unacceptably high dose rates associated with normal operation and maintenance of the SSCs in these systems.

The safety of the Japanese ABWR reference design of the systems described in this chapter is well understood, using proven technology. Hence a significant aspect of demonstrating the application of RGP in the design of the RCSs, the Reactivity Controls Systems and the Associated Systems of the UK ABWR is to generally adopt the Japanese reference design, with only limited modifications to reduce the risk So far As is Reasonably Practicable (SFAIRP) where it is reasonably practicable to do so as required by UK regulations. Thus, the UK ABWR design of these systems is mainly the same as for the operating ABWRs in Japan that is based in the previous BWRs designs, which together have provided many years of operating experience.
This implementation of RGP based on learning from experience has contributed to the reduction of residual risks compared with earlier ABWRs. However, on its own, it is not sufficient to fully demonstrate that the design meets the ALARP principle. This has been achieved as described below for the SSCs scope of this chapter.

The discussion on ALARP is generally included within each of the BSCs and TRs that support this chapter. Each BSC or TR also contains a section which discusses the following:

- With regard to design input, the safety functions that the SSCs have to deliver are linked to the fundamental safety functions, high level safety functions and fault schedule, and SFCs are set at a system level. Moreover, with regard to the performance required for the SSCs to deliver these safety functions, design requirements from the safety analysis, etc. are set as well.

- With regard to the compliance with regulations, RGP and OPEX, the requirements have been identified through collaboration with the UK experts such as the future licensee. The mature reference design with operational experience that the previous ABWR represents have been subjected to a gap analysis to make sure it is in line with UK practice.
- With regard to the gaps found between the reference ABWR design and UK regulations, RGP, etc., the UK ABWR SSCs design has gone through an optioneering and design review process to determine which options are the most reasonably practicable to close the gaps. Finally, the design was modified if appropriate.
- Hence, UK ABWR concept design specifications are set. Finally, the UK ABWR design that results from this process (modified reference design) is subjected to various risk assessments to make sure that the design reduces the risks (nuclear safety, worker safety and environmental impact) SFAIRP and is ALARP.

For the systems covered by Chapter 12, specific areas where RGP has been identified and applied in addition to the Japanese ABWR reference design include:

- ASME codes and standards,
- IAEA Safety Standards,
- WENRA Safety Objectives,
- Compliance with UK guidance on the design of safe isolation of plant and equipment for Examination, Maintenance, Inspection and Testing (EMIT) (Ref 12.5-3),
- Relevant good practice examples of other nuclear power plants, and
- Operational experience of other nuclear power plants.

For the systems covered by Chapter 12, specific areas where ALARP assessments have been used to inform the design and propose changes on the reference design include the following relevant examples:

- The reference design was subjected to a thorough assessment in order to demonstrate that the current isolation configuration of the mechanical systems reduces the risk as low as reasonably practicable when performing EMIT tasks. The assessment is still ongoing and several design changes were identified. For further details refer to the 'Strategy on Examination, Inspection Maintenance and Testing (EIM&T) Isolations and Configurations' (Ref 12.5-5).

  In relation to this topic, the reference design of the FMCRD was subjected to an optioneering and risk assessment to evaluate whether the isolation design reduced risk so far as reasonably practicable when performing EMIT tasks on the upper component of the FMCRD. The assessment concluded that a modification of the CRD handling machine was the most reasonably practicable thing to do. For further details refer to the 'ALARP Assessment Report for Fine Motion Control Rod Drive (FMCRD) Upper Component Maintenance' (Ref 12.5-6).

  From the same point of view as the FMCRD, the RIP reference design was subjected to an optioneering and risk assessment to evaluate whether the isolation design reduced risk so far as reasonably practicable when performing EMIT tasks. The assessment concluded that a modification of the RIP upper plug was the most reasonably practicable thing to do. For further details refer to the 'Optioneering Report for Maintenance of Reactor Internal Pump (RIP) Secondary Seal' (Ref 12.5-7).

- The reference design of the ME SSCs was subjected to a thorough assessment in order to demonstrate that their design life and replacement frequency are in line with RGP and reduce the risk as low as reasonably practicable. According to the design review process established, depending on the operational period and profile under which the SSCs are required to work for UK ABWR (construction and/or commercial operation and/or decommissioning) it is judged whether the SSC is provided with a design life of 60 years or longer, or periodically replaced whichever the most reasonably practicable approach is to ensure continued availability of the SSC for the required conditions. This assessment is still ongoing and the design life and replacement frequency of several SSCs have already been modified. For further details refer to section 5 of the 'Hitachi-GE Strategy on the Design Life of ME SSCs' (Ref 12.5-8).
  Specifically, for those SSCs that their availability is expected for operation during decommissioning, this safety demonstration must cover both commercial operation and decommissioning operational profiles showing that safety and environmental performance can be maintained based on a component/subsystem replacement policy. Where it is not reasonably practicable to replace a major structure or component then the future safety case and specification of requirements for an SSC must cover a full design life longer than 60 years and the differing operational profiles required for different phases of the overall lifecycle of the structure or component. See Chapter 31: Decommissioning, Section 31.5.2.4 for a list of the systems supporting decommissioning.

- The reference design of all ME SSCs was subjected to a thorough optioneering and risk assessment in order to demonstrate that the piping gradients are in line with relevant RGP and reduce the risk as low as reasonably practicable. As a result, Hitachi-GE policy on piping gradient was modified, considering the application of different gradients case by case. Based on these new principles, the piping gradient of all SSCs is being reviewed and modified when necessary. For further details refer to the 'ALARP Assessment for Piping Gradient' (Ref 12.5-9).

- The reference design of the CUW bottom drain line was subjected to an optioneering and risk assessment to determine whether it is ALARP or not to retain this line. This was triggered by international operational experience that has shown that this line can make a high contribution to worker dose during outages and is also a potential source of small LOCA. As a result of the ALARP evaluation process, it was concluded that the most reasonably practicable option was to retain the current bottom drain line. Details are described in the report 'ALARP Consideration on RPV Bottom Drain Line' (Ref 12.5-10).
  The reference design of the CUW Reactor Pressure Vessel Head Spray line was also subjected to an optioneering and risk assessment to determine whether it is ALARP or not to retain it. As a result of the ALARP evaluation process, it was concluded that the most reasonably practicable option was to retain the current head spray line. Details are described in the report 'ALARP Assessment for the Reactor Vessel Head Spray' (Ref 12.5-13).

- The selection of materials for the SSCs scope of this chapter was optimised to reduce risks of degradation and to minimise dose rates to levels that are ALARP. For further details refer to the 'Material Selection Report' (Ref 12.5-11) and related specific reports on the systems scope of this chapter such as the 'Material Selection Report for Nuclear Boiler Systems' (Ref 12.5-12).

- The reference design of the SRV at system and component levels was subjected to an optioneering and risk assessment to determine whether it is ALARP or not to provide further diversity in the design. This was especially triggered by the assessment of the design from the point of view of robustness against CCF and the consideration of international operational experiences that have shown that there are alternative diverse means compared to the current design of UK ABWR. As a result of the ALARP evaluation process, it was concluded that the current design of the SRV at both system and component levels was the most reasonably practicable option for the UK ABWR. Nonetheless, it was noted that the improvement of the design against CCF by the application of administrative measures such as diversity in the manufacturing process or maintenance tasks, etc. could potentially reduce the risk of CCF due to human errors, which turned out to be greater than the risk from the mechanical design itself. Therefore, it was recommended to further explore the application of diversity from these points of view post-GDA since they closely depend on future licensee. Details are described in the ‘Topic Report on Safety Relief Valve Diversity’ (Ref 12.5-4).

In summary, it is concluded that all reasonably practicable risk reduction measures have been implemented by the application of UK and international good practice and a systematic and comprehensive ALARP evaluation embedded within the design process. The design of the UK ABWR SSCs within the scope of Chapter 12 achieves risks levels that are therefore considered acceptable and ALARP.

12.6 Assumptions, Limits and Conditions for Operation

12.6.1 Purpose

This section considers the LCOs that apply specifically to the RCSs, the Reactivity Control Systems and the Associated Systems of the generic UK ABWR design reference.
The context, definition and process for identification of assumptions are described in Chapter 4: Safety Management throughout Plant Lifecycle.
The details are in the corresponding section of the BSC or TR of the related SSC:

RCSs and Associated Systems -

- RPV: (Ref 12.3-26)
- Reactor internal structures: (Ref 12.3-25)
- RCPB Overpressure Protection System: (Ref 12.3-1)
- LDS: (Ref 12.3-2)
- RRS: (Ref 12.3-3]
- NB: (Ref 12.3-4)
- CUW: (Ref 12.3-5)
- RHR: (Ref 12.3-6)

Description of Reactivity Control Systems -

- CRD: (Ref 12.4-1)
- SLC: (Ref 12.4-2]

12.6.2 LCOs specified for RCSs, Reactivity Control Systems and Associated Systems

The LCOs that apply to the RCSs, the Reactivity Control Systems and the Associated Systems are identified under each individual system within the chapter.

12.6.3 Assumptions for RCSs, Reactivity Control Systems and Associated Systems

As mentioned in section 12.1.2, there are no fundamental assumptions related to the SSCs scope of this chapter.

12.7 Conclusions

This chapter and its supporting references provide a demonstration that the RCSs, the Reactivity Control Systems and the Associated Systems for the UK ABWR have been designed to the highest standards consistent with their safety critical role for the UK ABWR.

Section 12.3.5 has demonstrated that a key innovation over many earlier generation BWRs is the use in the RRS of RIPs. RIPs introduced a major safety innovation of eliminating the connection of pipes to the RPV below the core by removing the need to connect external recirculation piping.

An important demonstration provided in this chapter is the high integrity design of the NB, the analysis described in this chapter also depends on the structural integrity methods described in Chapter 8: Structural Integrity and it supporting references. As well as describing the very high quality design of the NB and its internal components this chapter also demonstrates the considerable defence in depth of the system of 16 SRVs. This system has considerable redundancy and it also provides seven SRVs to perform the ECCS ADS safety function and another seven of SRVs (14 out of the 16 SRVs) performing the RDCF safety function (for details of the ECCS see Chapter 13: Engineered Safety Features). MSIVs are a critical set of components of the NB. This chapter has demonstrated that the inboard MSIVs are designed as VHI components (see Chapter 8: Structural Integrity for additional information) and have low spurious failures rates while being able to close rapidly with very high reliability when initiated by the Safety Class 1 C&I System (see Chapter 14: Control and Instrumentation for additional information).

An important element of successful low radiation environment reactor operations is to retain high quality water by the removal of impurities. This chapter and its supporting references have demonstrated the effectiveness of the CUW to ensure a low corrosion and low radioactive contamination environment throughout the operational life of the facility.

Following the successful shutdown of the reactor through the control rods heat is still generated through the continuing radioactive decay of the fission products; this is known as residual heat. This chapter has described the role of the RHR in ensuring the provision of a high integrity system for cooling the residual heat. The RHR is a highly reliable Safety Class 1 system with three fully segregated divisions of equipment only one of which is required to ensure the successful cooling of the residual heat generated within the reactor. This chapter also describes the multi-functional role of the RHR acting, for example, as the LPFL for the ECCS (see Chapter 13) and providing cooling for removal of heat from the PCV, etc.

There are two Reactivity Control Systems, the CRD and the SLC. This chapter and its supporting references describe the safety critical role of these two systems and how their designs have been optimised to ensure highly reliable and fault tolerant operation. For example, on the CRD, it describes the use of 103 HCUs to provide high speed injection of 205 control rods into the reactor on the initiation of a signal to scram (rapid shutdown of reactivity) the reactor. The 103 HCUs can be scrammed from either the Safety Class 1 SSLC or from the Safety Class 2 HWBS. Despite the high level of fault tolerance provided by the CRD this chapter shows that the diverse SLC effectively shuts down the reactor with the injection of a sodium pentaborate solution into the reactor in the highly unlikely event of a CCF of the CRD.

A key element of this chapter and its supporting references has been to show that the design of the UK ABWR RCSs, Reactivity Control Systems and Associated Systems reduce the risks of reactor accidents to a level that is as low as is reasonably practicable.

NOT PROTECTIVELY MARKED

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Appendix A: Safety Functional Claims Tables

The Safety Functional Claims table of each system is provided in the following tables.

Section 12.3

- SFC Table of RCPB
- SFC Table of RCPB Overpressure Protection System
- SFC Table of RCPB Leakage Detection System
- SFC Table of RPV
- SFC Table of RIN
- SFC Table of RRS
- SFC Table of NB
- SFC Table of CUW
- SFC Table of RHR
- SFC Table of VGL

Section 12.4

- SFC Table of CRD
- SFC Table of SLC

SFC Table of RCPB

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[RRS SFC 4-1.1]	The RRS portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
2	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[NB SFC 4-1.1]	The NB portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
3	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[CUW SFC 4-1.1]	The CUW portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
4	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[RHR SFC 4-1.1]	The RHR portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
5	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[RCIC SFC 4-1.1]	The RCIC portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
6	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[HPCF SFC 4-1.1]	The HPCF portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
7	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[CRD SFC 4-1.1]	The CRD portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
8	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[SLC SFC 4-1.1]	The SLC portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1

SFC Table of RCPB Overpressure Protection System

	Top Claim for Mechanical System							Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)							
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)			State	Claim ID	Claim Contents	Cat.	Class
1	2	Fuel Cooling	2-1	Functions to cool reactor core	1.1 1.4 1.7 1.8 2.1 2.2 2.3 3.1 4.6 5.1 5.2 5.3 6.1 10.1 10.2 10.3 11.3 11.5 11.8 11.9 11.10 11.11 11.12 17.1 17.2 17.3 17.4 17.5 17.6 18.1 18.2 18.3	Generator load rejection FDW controller failure - Maximum demand Reactor pressure regulator failure in the closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all FDW flow Radiation monitor failure Short term LOOP Medium term LOOP Long term LOOP Inadvertent opening of a SRV LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break Inadvertent start-up all injection system Inadvertent MSIV closure due to spurious failure of Class 1 SSLC M/C power supply failure on electrical CCF D/C power supply function failure on electrical CCF Loss of all RCW Loss of all RSW Loss of Class 1 HVAC Internal Fire in the Reactor Building Internal Fire in the Heat Exchanger Building Internal Fire in the Control Building Internal Fire in the Main Control Room Internal Missile in the Main Control Room Turbine Missile Loss of Ultimate Heat Sink 10^{-3}/year Earthquake Design Basis Earthquake (DBE)		Fault Conditions	[NB SFC 2-1.1]	The MS through the safety valve function of the SRV is the principal means to release the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV.	A	1
2	2	Fuel Cooling	2-1	Functions to cool reactor core	-	No claim for the design basis		Fault Conditions	[NB SFC 2-1.2]	The MS through the relief valve function of the SRV is an additional means to the release of the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV.	C	3

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
3	4	Confinement/Containment of radioactive materials	4-2	Functions to prevent overpressure within the reactor coolant pressure boundary	The same faults as [NB SFC 2-1.1]		Fault Conditions	[NB SFC 4-2.1]	The MS through the safety valve function of the SRVs is the principal means to deliver overpressure protection of the RCPB under abnormal transients and accident conditions that could put excessive pressure on the boundary.	A	1
4	4	Confinement/Containment of radioactive materials	4-5	Functions to reseat safety valves and relief valves	The same faults as [NB SFC 4-2.1]		Fault Conditions	[NB SFC 4-5.1]	The MS is a principal means to prevent excessive loss of reactor coolant after SRV actuation for the safety valve function.	B	3
5	4	Confinement/Containment of radioactive materials	4-6	Functions to mitigate reactor pressure increase with other system (other than No. 4-2)	-	No claim for the design basis	Fault Conditions	[NB SFC 4-6.1]	The MS through the relief valve function of the SRVs is an additional means to deliver overpressure protection of the RCPB faster than the safety valve function under abnormal transients and accident conditions that could put excessive pressure on the boundary.	C	3

SFC Table of RCPB Leakage Detection System

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	4	Confinement/Containment of radioactive materials	4-3	Functions to contain reactor coolant outside the RCPB	-	No corresponding fault	Normal and Fault Conditions	[LDS SFC 4-3.1]	The LDS is a principal means to deliver monitoring of leakage within the reactor coolant system.	C	3
2	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW LOCA - MS line break LOCA - RHR outlet line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FDW line (RCIC connected) break	Fault Conditions	[LDS SFC 4-7.1]	The LDS is a principal means to deliver detection and alarm of leakage from the reactor coolant system as well as initiation of the signals to isolate the corresponding systems in the event a leakage is detected.	A	1

SFC Table of RPV

	Top Claim for Mechanical Components						Safety Functional Claim for the Mechanical Systems and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-2	Functions to maintain core geometry	-	No Claim	Normal and Fault Conditions	[RPV SFC 1-2.1]	The RPV supports the core support structure and other reactor internals.	A	1
2	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	-	No Claim	Fault Conditions	[RPV SFC 1-3.1]	The RPV supports and aligns the CRD and In-Core flux monitoring (ICM) assemblies.	A	1
3	2	Fuel Cooling	2-1	Functions to cool reactor core	-	No Claim	Normal and Fault Conditions	[RPV SFC 2-1.1]	The RPV, in conjunction with the core support structure and other reactor internals, allows for flow of coolant through the reactor core.	A	1
4	3	Long term heat removal	3-1	Functions to remove residual heat after shutdown	-	No Claim	Normal and Fault Conditions	[RPV SFC 3-1.1]	The RPV assure the structural integrity to maintain the reactor coolant volume to ensure core cooling or residual heat capabilities.	A	1
5	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Normal and Fault Conditions	[RPV SFC 4-1.1]	The RPV provides a pressure boundary to contain the reactor coolant, nuclear fuel and fission products for all design basis conditions over 60 years.	A	1
6	5	Others	5-1	Functions to generate actuation signals for the engineered safety features and reactor shutdown systems	-	No Claim	Normal and Fault Conditions	[RPV SFC 5-1.1]	The RPV supports and aligns its connected pipework and instrumentation lines.	A	1

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SFC Table of RIN

	Top Claim for Mechanical Component						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1Fault Schedule(Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-2	Functions to maintain core geometry	-	No Claim	Normal and Fault Conditions	[RIN SFC 1-2.1]	Reactor Internals assure their structural integrity in order to maintain core geometry.	A	1
2	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	-	No Claim	Fault Conditions	[RIN SFC 1-3.1]	Reactor Internals assure their structural integrity in order to provide the guide for the rapid insertion of Control Rod to achieve sub-criticality.	A	1
3	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	-	No Claim	Fault Conditions	[RIN SFC 1-4.1]	Reactor Internals assure their structural integrity in order to inject the sodium pentaborate solution into the core to achieve and maintain sub-criticality by SLC in case of CRs insertion failure.	A	2
4	2	Fuel Cooling	2-1	Functions to cool reactor core	-	No Claim	Fault Conditions	[RIN SFC 2-1.1]	Reactor Internals assure their structural integrity in order to form appropriate passage for coolant flow.	A	1
5	3	Long term heat removal	3-1	Functions to remove residual heat after shutdown	-	No Claim	Fault Conditions	[RIN SFC 3-1.1]	Reactor Internals assure their structural integrity in order to form appropriate flow path for residual heat removal system.	A	1
6	5	Others	5-1	Functions to generate actuation signals for the engineered safety features and reactor shutdown systems	-	No Claim	Fault Conditions	[RIN SFC 5-1.1]	Reactor Internals assure their structural integrity in order to actuate signals for the reactor shutdown system.	A	1

SFC Table of RRS

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-6	Functions to circulate reactor coolant (functions to control reactivity of the core in normal operational states)	-	No corresponding fault	Normal Conditions	[RRS SFC 1-6.1]	The RRS in conjunction with the RFC provides reactor coolant forced recirculation for power generation during normal operation conditions, whose failure could lead to a total loss of reactor coolant flow and consequently demand a Category A safety function to mitigate it.	B	3
2	1	Control of Reactivity	1-6	Functions to circulate reactor coolant (functions to control reactivity of the core in normal operational states)	-	No corresponding fault	Normal Conditions	[RRS SFC 1-6.2]	The RRS in conjunction with the RFC provides reactor coolant forced recirculation for power generation during normal operation conditions, whose failure could lead to a partial loss of reactor coolant flow and consequently demand a Category A safety function to mitigate it depending on the case.	B	3
3	1	Control of Reactivity	1-5	Functions of alternative reactivity control	1.1 1.4 1.5 1.7 1.8 2.1 2.2 2.3 4.2 17.1 17.2 17.3 17.4 17.6 18.1 18.2	Generator load rejection FDW controller failure - Maximum demand Recirculation flow control failure (Runout of all reactor internal pumps) Reactor pressure regulator failure in the closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Control rod withdrawal error at power Internal Fire in the Reactor Building Internal Fire in the Heat Exchanger Building Internal Fire in the Control Building Internal Fire in the Main Control Room Turbine Missile Loss of Ultimate Heat Sink 10^{-3}/year Earthquake	Fault Conditions	[RRS SFC 1-5.1]	The RIPs of the RRS are tripped by the RPT by a signal from the Hardwired Backup System (HWBS) as part of the actions to perform alternative shutdown of the reactor in the event of Anticipated Transient Without Scram (ATWS).	A	2(3)
4	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system		No claim for the design basis	Fault Conditions	[RRS SFC 1-8.1]	The RIPs of the RRS are tripped by the RPT by a signal from the Plant Control System (PCS) as part of the actions used to deliver mitigation of power increases.	C	3
5	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[RRS SFC 4-1.1]	The RRS portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
6	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	[RRS SFC 4-7.1]	The RRS components penetrating the primary containment form a barrier to confine radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1

SFC Table of NB

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 : Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-5	Functions of alternative reactivity control	1.1 1.4 1.5 1.7 1.8 2.1 2.2 2.3 4.2 17.1 17.2 17.3 17.4 17.6 18.1 18.2	Generator load rejection FDW controller failure - Maximum demand Recirculation flow control failure (Runout of all reactor internal pumps) Reactor pressure regulator failure in the closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Control rod withdrawal error at power Internal Fire in the Reactor Building Internal Fire in the Heat Exchanger Building Internal Fire in the Control Building Internal Fire in the Main Control Room Turbine Missile Loss of Ultimate Heat Sink 10^{-3}/year Earthquake	Fault Conditions	[NB SFC 1-5.1]	The FDW flow is controlled by the Hardwired Back-up System logic in order to prevent reactivity insertion in the event of Anticipated Transient Without Scram (ATWS).	A	2
2	2	Fuel Cooling	2-1	Functions to cool reactor core	1.1 1.4 1.7 1.8 2.1 2.2 2.3 3.1 4.6 5.1 5.2 5.3 6.1 10.1 10.2 10.3 11.3 11.5 11.8 11.9 11.10	Generator load rejection FDW controller failure - Maximum demand Reactor pressure regulator failure in the closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all FDW flow Radiation monitor failure Short term LOOP Medium term LOOP Long term LOOP Inadvertent opening of a SRV LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break Inadvertent start-up all injection system Inadvertent MSIV closure due to spurious failure of Class 1 SSLC M/C power supply failure on electrical CCF D/C power supply function failure on electrical CCF Loss of all RCW	Fault Conditions	[NB SFC 2-1.1]	The MS through the safety valve function of the SRV is the principal means to release the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 : Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
					11.11 11.12 17.1 17.2 17.3 17.4 17.5 17.6 18.1 18.2 18.3	Loss of all RSW Loss of Class 1 HVAC Internal Fire in the Reactor Building Internal Fire in the Heat Exchanger Building Internal Fire in the Control Building Internal Fire in the Main Control Room Internal Missile in the Main Control Room Turbine Missile Loss of Ultimate Heat Sink 10^{-3}/year Earthquake Design Basis Earthquake (DBE)					
3	2	Fuel Cooling	2-1	Functions to cool reactor core	-	No claim for the design basis	Fault Conditions	[NB SFC 2-1.2]	The MS through the relief valve function of the SRV is an additional means to the release of the steam generated during reactor core cooling by high pressure core cooling systems in the event of faults such as LOCA outside the PCV.	A	1
4	2	Fuel Cooling	2-1	Functions to cool reactor core	7.1 8.1 8.2 9.1 9.2 9.3	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line LOCA - MS line break LOCA - RHR line break	Fault Conditions	[NB SFC 2-1.3]	The NB through the ADS is the principal means to depressurise the RPV in order to provide reactor core cooling in low pressure state as part of the ECCS so that significant damage to the fuel is prevented and the reaction between the fuel cladding and the reactor coolant is sufficiently minimised in the event of LOCA inside the PCV.	A	1
5	2	Fuel Cooling	2-1	Functions to cool reactor core	10.1 10.2 10.3	LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break	Fault Conditions	[NB SFC 2-1.4]	The NB through the transient ADS is the principal means to depressurise the RPV in order to provide reactor core cooling in low pressure state as part of the ECCS so that significant damage to the fuel is prevented and the reaction between the fuel cladding and the reactor coolant is sufficiently minimised in the event of LOCA outside the PCV	A	1
6	3	Long term heat Removal	3-1	Functions to remove residual heat after shutdown	1.1 1.2 1.3 1.4	Generator load rejection Partial loss of reactor coolant flow Loss of reactor coolant flow (Trip of all reactor internal pumps) FDW controller failure – Maximum demand	Fault Conditions	[NB SFC 3-1.1]	The MS through the SRVs is the principal means to deliver long-term residual heat removal to reach reactor cold shutdown by depressurisation of the RPV in the event of	A	1

	Top Claim for Mechanical System				Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)	High Level Safety Function (HLSF)	Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR	Topic Report on Fault Assessment Table.4.2-1 : Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
			1.5	Recirculation flow control failure (Runout of all reactor internal pumps)			unavailability of the main condenser		
			1.6	Loss of FDW heating					
			1.8	Inadvertent control valve closure					
			2.1	Inadvertent MSIV closure					
			2.2	Reactor pressure regulator failure in the open direction					
			2.3	Loss of main condenser vacuum					
			3.1	Loss of all FDW flow					
			4.2	Control rod withdrawal error at power					
			4.4	Inadvertent rector scram [CRD pump trip]					
			4.5	SRNM or APRM sensor failure					
			4.6	Radiation monitor failure					
			5.1	Short term LOOP					
			5.2	Medium term LOOP					
			5.3	Long term LOOP					
			6.1	Inadvertent opening of a SRV					
			7.2	Small line break LOCA					
			10.1	LOCA outside primary containment –Main steam line Break					
			10.2	LOCA outside primary containment –Reactor water clean-up line break					
			10.3	LOCA outside primary containment –FDW line (RCIC connected) break					
			10.4	Small line break LOCA outside primary containment					
			11.3	Inadvertent start-up all injection system					
			11.5	Inadvertent MSIV closure due to spurious failure of Class 1 SSLC					
			11.8	M/C power supply failure on electrical CCF					
			11.10	Loss of all RCW					
			11.11	Loss of all RSW					
			11.12	Loss of Class 1 HVAC					
			17.1	Internal Fire in the Reactor Building					
			17.2	Internal Fire in the Heat Exchanger Building					
			17.3	Internal Fire in the Control Building					
			17.4	Internal Fire in the Main Control Room					
			17.5	Internal Missile in the Main Control Room					
			17.6	Turbine Missile					
			18.1	Loss of Ultimate Heat Sink					
			18.2	10^{-3}/year Earthquake					
			18.3	Design Basis Earthquake (DBE)					

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 : Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
7	3	Long term heat Removal	3-1	Functions to remove residual heat after shutdown	13.1 13.3 13.4 13.5 13.6 13.7 13.12	Loss of reactor coolant flow Loss of operating RHR with loss of the same division of ECCS Loss of operating RHR due to CCF of Class 1 controller Loss of off-site power SBO Draindown due to valve failure within the operating RHR RPV draindown by CUW	Fault Conditions	[NB SFC 3-1.2]	The MS through the SRVs is the principal means to deliver long-term residual heat removal during shutdown in the event of unavailability of the RHR.	A	1
8	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[NB SFC 4-1.1]	The NB portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
9	4	Confinement/Containment of radioactive materials	4-2	Functions to prevent overpressure within the reactor coolant pressure boundary	The same faults as [NB SFC 2-1.1]		Fault Conditions	[NB SFC 4-2.1]	The MS through the safety valve function of the SRVs is the principal means to deliver overpressure protection of the RCPB under abnormal transients and accident conditions that could put excessive pressure on the boundary.	A	1
10	4	Confinement/Containment of radioactive materials	4-3	Functions to contain reactor coolant outside the RCPB	-	No corresponding fault	Normal Conditions	[NB SFC 4-3.1]	The MS pipework of the NB outside the RCPB (beyond outboard MSIV) contains reactor coolant and its rupture could lead to a release of radioactive material of dose consequences relatively low, but demanding Category A safety functions to mitigate them.	B	3
11	4	Confinement/Containment of radioactive materials	4-3	Functions to contain reactor coolant outside the RCPB	-	No corresponding fault	Normal Conditions	[NB SFC 4-3.2]	The FDW pipework of the NB outside the RCPB (beyond the outboard check valve) contains reactor coolant and its rupture could lead to a release of radioactive material of dose consequences relatively low, but demanding Category A safety functions to mitigate them.	B	3
12	4	Confinement/Containment of radioactive materials	4-5	Functions to reseat safety valves and relief valves	The same faults as [NB SFC 4-2.1]		Fault Conditions	[NB SFC 4-5.1]	The MS is a principal means to prevent excessive loss of reactor coolant after SRV actuation for the safety valve function.	B	3

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 : Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
13	4	Confinement/Containment of radioactive materials	4-6	Functions to mitigate reactor pressure increase with other systems (other than No. 4-2)	-	No claim for the design basis	Fault Conditions	[NB SFC 4-6.1]	The MS through the relief valve function of the SRVs is an additional means to deliver overpressure protection of the RCPB faster than the safety valve function under abnormal transients and accident conditions that could put excessive pressure on the boundary.	C	3
14	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	10.1	LOCA outside PCV - MS line break	Fault Conditions	[NB SFC 4-7.1]	The MS through its flow restrictors is a principal means to limit the loss of reactor coolant and the release of radioactive material from the RPV following a MS line rupture outside the PCV to the extent that the RPV water level does not drop below the top of the active fuel before closure of the MSIVs.	A	1
15	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	2.1 2.2 2.3 3.1 4.6 5.1 5.2 5.3 7.1 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3 11.2 11.4 11.8 11.9 11.10 11.11 11.12 17.6 18.1 18.2	Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all FDW flow Radiation monitor failure Short term LOOP Medium term LOOP Long-term LOOP LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line (LPFL connected) break LOCA - MS line break LOCA - RHR line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break Inadvertent opening of all ADS Inadvertent opening of all ADS due to spurious failure of Class 1 SSLC M/C power supply failure on electrical CCF D/C power supply Function failure on electrical CCF Loss of all RCW Loss of all RSW Loss of Class 1 HVAC Turbine Missile Loss of Ultimate Heat Sink 10^{-3}/year Earthquake	Fault Conditions	[NB SFC 4-7.2]	The MS is the principal means to close the MS lines to limit the release of reactor coolant and radioactive material to the surroundings in the event of a MS pipe rupture by closing the MSIVs.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 : Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
					18.3	Design Basis Earthquake (DBE)					
16	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line (LPFL connected) break (9.1.1, 9.1.2) LOCA - MS line break LOCA - RHR line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break	Fault Conditions	[NB SFC 4-7.3]	The NB components penetrating the primary containment form a barrier to confine radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1
17	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 2.1 2.2 2.3 3.1 4.2 4.4 4.5 4.6 5.1 5.2 5.3 6.1 7.2 10.1 10.2 10.3 10.4 11.3 11.5 11.8 11.9 11.10	Generator load rejection Partial loss of reactor coolant flow Loss of reactor coolant flow (Trip of all reactor internal pumps) FDW controller failure - Maximum demand Recirculation flow control failure (runout of all RIPs) Loss of FDW heating Reactor pressure regulator failure in the closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all FDW flow Control rod withdrawal error at power Inadvertent rector scram [CRD pump trip] SRNM or APRM sensor failure Radiation monitor failure Short term LOOP Medium term LOOP Long-term LOOP Inadvertent opening of a SRV Small line break LOCA LOCA outside primary containment –Main steam line break– LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break Small line break LOCA outside primary containment Inadvertent start-up all injection system Inadvertent MSIV closure due to spurious failure of Class 1 SSLC M/C power supply failure on electrical CCF D/C power supply failure on electrical CCF Loss of all RCW	Fault Conditions	[NB SFC 4-7.4]	The MS through the SRV discharge line quenchers is a principal means to suppress the dynamic loads generated in the PCV when steam discharged via the SRVs condenses in the suppression pool.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 : Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
					11.11	Loss of all RSW					
					11.12	Loss of Class 1 HVAC					
					13.3	Loss of operating RHR with loss of the same division of ECCS					
					13.4	Loss of operating RHR due to CCF of Class 1 controller					
					13.5	Loss of off-site power					
					13.6	SBO					
					13.7	Draindown due to valve failure within the operating RHR					
					13.12	RPV draindown by CUW					
					17.1	Internal Fire in the Reactor Building					
					17.2	Internal Fire in the Heat Exchanger Building					
					17.3	Internal Fire in the Control Building					
					17.4	Internal Fire in the Main Control Room					
					17.5	Internal Missile in the Main Control Room					
					17.6	Turbine Missile					
					18.1	Loss of Ultimate Heat Sink					
					18.2	10^{-3}/year Earthquake					
					18.3	Design Basis Earthquake (DBE)					

SFC Table of CUW

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	5	Others	5-8	Functions to clean up reactor coolant	-	No corresponding fault	Normal Conditions	[CUW SFC 5-8.1]	The CUW provides a continuous purifying treatment of reactor water by removing soluble and insoluble impurities during normal operation.	C	3
2	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[CUW SFC 4-1.1]	The CUW portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
3	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line (9.1.1, 9.1.2) LOCA - MS line break LOCA - RHR line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break	Fault Conditions	[CUW SFC 4-7.1]	The CUW components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1
4	4	Confinement/Containment of radioactive materials	4-3	Functions to contain reactor coolant outside the RCPB	-	No corresponding fault	Normal Conditions	[CUW SFC 4-3.1]	The CUW piping and components outside the RCPB contain reactor coolant. A breach could lead to a release of radioactive material of dose consequences that are relatively low, but demanding Safety Category A safety functions to mitigate them.	B	3

SFC Table of RHR

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	SFC Contents	Cat.	Class
1	3	Long Term Heat Removal	3-1	Functions to remove residual heat after shutdown	-	No corresponding fault	Normal Conditions	[RHR SFC 3-1.1]	The RHR through its Reactor Shutdown Cooling mode is the principal means to remove residual heat after normal reactor shutdown to reach reactor cold shutdown.	A	1
2	2	Fuel Cooling	2-4	Function to cool spent fuel outside the reactor coolant system	-	No corresponding fault	Normal Conditions	[RHR SFC 2-4.1]	The RHR provides the FPC with supplemental cooling to maintain the SFP water temperature within the design values by removing decay heat in the event of full core offload where the heat load to the pool exceeds the FPC cooling capacity. This function can also be used for recovery from potential upper pools cooling failure and subsequent boiling event.	A	1
3	4	Confinement/ Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[RHR SFC 4-1.1]	The RHR portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1
4	4	Confinement/ Containment of radioactive materials	4-3	Functions to contain reactor coolant outside the RCPB	-	No corresponding fault	Normal Conditions	[RHR SFC 4-3.1]	When the RHR is operating in Shutdown Cooling mode, its piping and components outside the RCPB contain reactor coolant. A breach could lead to a release of radioactive material of dose consequences relatively low, but demanding Safety Category A safety functions to mitigate them.	B	3
5	4	Confinement/ Containment of radioactive materials	4-4	Functions to contain radioactive material	-	No corresponding fault	Normal Conditions	[RHR SFC 4-4.1]	When the RHR is in standby, its piping and components contain material with low radioactivity. A breach could lead to a release of radioactive material of dose consequences relatively low.	C	3
6	2	Fuel Cooling	2-1	Functions to cool reactor core	7.1 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3 13.8 13.9 13.10 13.11	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line LOCA - MS line break LOCA - RHR line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line break SD - LOCA - feedwater line inside PCV SD - LOCA - RHR suction line inside PCV SD - LOCA - LPFL return line inside PCV SD - LOCA -mechanical below TAF	Fault Conditions	[RHR SFC 2-1.1]	The LPFL is a principal means to provide reactor core cooling as part of the ECCS when the RPV is in low pressure state so that significant damage to the fuel is prevented and the reaction between the fuel cladding and the reactor coolant is minimised in the event of infrequent faults such as LOCA.	A	1
7	3	Long term heat removal	3-1	Functions to remove residual heat after shutdown	1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8	Generator load rejection Partial loss of reactor flow (trip of 4 RIPs) Loss of reactor flow (trip of all RIPs) Feedwater controller failure - Maximum demand Recirculation flow control failure (runout of all RIPs) Loss of feedwater heating Reactor pressure regulator failure in the closed direction Inadvertent control valve closure	Fault Conditions	[RHR SFC 3-1.2]	The RHR through its Reactor Shutdown Cooling mode is a principal means to deliver long term containment heat removal by removing the decay heat of fission products from the reactor without exceeding the fuel design margins and RCPB design conditions after reactor shutdown following frequent faults such as main condenser unavailability, and infrequent faults such as SBO after power recovery.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	SFC Contents	Cat.	Class
					2.1 2.2 2.3 3.1 4.2 4.4 4.5 4.6 5.1 5.2 5.3 6.1 7.2 10.1 10.2 10.3 10.4 11.2 11.3 11.4 11.5 11.8 11.9 11.10 11.11 11.12 13.3 13.4 13.5 13.6 13.7 13.12 17.1 17.2 17.3 17.4 17.5 17.6 18.1 18.2 18.3	Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Control rod withdrawal error at power Inadvertent reactor SCRAM (CRD pump trip) SRNM or APRM sensor failure Radiation Monitor Failure Short term LOOP Medium term LOOP Long-term LOOP Inadvertent opening of a SRV LOCA - small line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line break LOCA outside PCV - small line break Inadvertent opening of all ADS Inadvertent start up of all injection system Inadvertent opening of all ADS (SSLC failure) Inadvertent MSIV closure (SSLC failure) M/C power supply failure on electrical CCF D/C power supply failure on electrical CCF Loss of all RCW Loss of all RSW Loss of Class 1 HVAC SD - Loss of operating RHR and the same ECCS div. SD – Loss of operating RHR (Class 1 controller CCF) SD - LOOP SD - SBO SD - Draindown due to operating RHR valve failure SD - RPV draindown by CUW Internal fire in R/B Internal fire in Hx/B Internal fire in C/B Internal fire in MCR Internal missile in MCR Turbine missile Loss of UHS 10^{-3}/y earthquake DB earthquake					
8	3	Long term heat removal	3-1	Functions to remove residual heat after shutdown	1.1 1.2 1.3	Generator load rejection Partial loss of reactor flow (trip of 4 RIPs) Loss of reactor flow (trip of all RIPs)	Fault Conditions	[RHR SFC 3-1.3]	The RHR through its Suppression Pool Cooling mode (SPC) is a principal means to deliver long-term containment heat removal following frequent faults such as main condenser unavailability	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	SFC Contents	Cat.	Class
					1.4	Feedwater controller failure - Maximum demand			and infrequent faults such as Anticipated Transient Without Scram (ATWS).		
					1.5	Recirculation flow control failure (runout of all RIPs)					
					1.6	Loss of feedwater heating					
					1.7	Reactor pressure regulator failure in the closed direction					
					1.8	Inadvertent control valve closure					
					2.1	Inadvertent MSIV closure					
					2.2	Reactor pressure regulator failure in the open direction					
					2.3	Loss of main condenser vacuum					
					3.1	Loss of all feedwater flow					
					4.2	Control rod withdrawal error at power					
					4.4	Inadvertent reactor SCRAM (CRD pump trip)					
					4.5	SRNM or APRM sensor failure					
					5.1	Short LOOP					
					5.2	Medium LOOP					
					5.3	Long LOOP					
					6.1	Inadvertent opening of a SRV					
					7.2	LOCA - small line break					
					10.1	LOCA outside PCV - MS line break					
					10.2	LOCA outside PCV - CUW line break					
					10.3	LOCA outside PCV - FWD line					
					10.4	LOCA outside PCV - small line break					
					11.3	Inadvertent start up of all injection system					
					11.4	Inadvertent opening of all ADS (SSLC failure)					
					11.5	Inadvertent MSIV closure (SSLC failure)					
					13.7	SD - Draindown due to operating RHR valve failure					
					13.8	SD - LOCA inside PCV – FWD line					
					13.9	SD - LOCA inside PCV – RHR suction line					
					13.10	SD - LOCA inside PCV – LPFL injection line					
					13.11	SD - LOCA below TAF					
					13.12	SD - RPV draindown by CUW					
					13.13	SD - Leakage during FMCRD inspection					
					13.14	SD - Leakage during IMC nozzle replacement					
					13.15	SD - Leakage during RIP inspection					
					13.16	SD - Refuelling bellow perforation					
					17.1	Internal fire in R/B					
					17.2	Internal fire in Hx/B					
					17.3	Internal fire in C/B					
					17.4	Internal fire in MCR					
					17.5	Internal missile in MCR					
					17.6	Turbine missile					
					18.1	Loss of UHS					
					18.2	10^{-3}/y earthquake					
					18.3	DB earthquake					
9	3	Long term heat removal	3-1	Functions to remove residual heat after shutdown	1.1 1.2	Generator load rejection Partial loss of reactor flow (trip of 4 RIPs)	Fault Conditions	[RHR SFC 3-1.4]	The RHR through its LPFL mode is a principal means to deliver long-term containment heat removal following frequent faults	A	1

	Top Claim for Mechanical System				Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)	High Level Safety Function (HLSF)	Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR	Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	SFC Contents	Cat.	Class
			1.3	Loss of reactor flow (trip of all RIPs)			such as main condenser unavailability and infrequent faults such as LOCA.		
			1.4	Feedwater controller failure - Maximum demand					
			1.5	Recirculation flow control failure (runout of all RIPs)					
			1.6	Loss of feedwater heating					
			1.7	Reactor pressure regulator failure in the closed direction					
			1.8	Inadvertent control valve closure					
			2.1	Inadvertent MSIV closure					
			2.2	Reactor pressure regulator failure in the open direction					
			2.3	Loss of main condenser vacuum					
			2.4	Radiation monitor failure					
			3.1	Loss of all feedwater flow					
			4.2	Control rod withdrawal error at power					
			4.4	Inadvertent reactor SCRAM (CRD pump trip)					
			4.5	SRNM or APRM sensor failure					
			5.1	Short LOOP					
			5.2	Medium LOOP					
			5.3	Long LOOP					
			6.1	Inadvertent opening of a SRV					
			7.1	LOCA - RPV bottom drain line break					
			7.2	LOCA - small line break					
			8.1	LOCA - HPCF line break					
			8.2	LOCA - LPFL line break					
			9.1	LOCA - FWD line					
			9.2	LOCA - MS line break					
			9.3	LOCA - RHR line break					
			10.1	LOCA outside PCV - MS line break					
			10.2	LOCA outside PCV - CUW line break					
			10.3	LOCA outside PCV - FWD line					
			10.4	LOCA outside PCV - small line break					
			11.2	Inadvertent opening of all ADS					
			11.3	Inadvertent start up of all injection system					
			13.3	SD- Loss of operating RHR and the same ECCS div.					
			13.5	SD - LOOP					
			13.7	SD - Draindown due to operating RHR valve failure					
			13.8	SD - LOCA inside PCV – FWD line					
			13.9	SD - LOCA inside PCV – RHR suction line					
			13.10	SD - LOCA inside PCV – LPFL injection line					
			13.11	SD - LOCA below TAF					
			13.12	SD - RPV draindown by CUW					
			13.13	SD - Leakage during FMCRD inspection					
			13.14	SD - Leakage during IMC nozzle replacement					
			13.15	SD - Leakage during RIP inspection					
			13.16	SD - Refuelling bellow perforation					
			17.1	Internal fire in R/B					
			17.2	Internal fire in Hx/B					

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	SFC Contents	Cat.	Class
					17.3 17.5 17.6 18.1 18.2 18.3	Internal fire in C/B Internal missile in MCR Turbine missile Loss of UHS 10^{-3}/y earthquake DB earthquake					
10	3	Long term heat removal	3-1	Functions to remove residual heat after shutdown	5.1 5.2 11.8 11.9 11.10 11.11 11.12 18.1	Short LOOP & CCF of EDGs Medium LOOP & CCF of EDGs M/C power supply failure (electrical CCF) D/C power supply failure (electrical CCF) Loss of all RCW Loss of all RSW Loss of Class 1 HVAC Loss of UHS	Fault Conditions	[RHR SFC 3-1.5]	The RHR through its Suppression Pool Cooling mode (SPC) is a principal means to deliver long term containment heat removal upon RHR recovery following venting during infrequent faults such as SBO.	A	1
11	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault.	Fault Conditions	[RHR SFC 4-7.2]	The PCV Spray Cooling mode of the RHR contributes to suppress PCV atmosphere pressure and remove fission products from the containment atmosphere during a LOCA inside PCV.	B	2
12	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line (9.1.1, 9.1.2) LOCA - MS line break LOCA - RHR line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break	Fault Conditions	[RHR SFC 4-7.1]	The RHR components penetrating the primary containment form barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1

SFC Table of VGL

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	4	Confinement/Containment of radioactive materials	4-12	Functions to store the radioactive materials as liquid wastes	-	No Claim	Normal Conditions	[VGL SFC 4-12.1]	The VGL collects reactor coolant leakage from the gland seal of the target valves and transfers it to the corresponding collection point during normal operation.	C	3
2	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line (9.1.1, 9.1.2) LOCA - MS line break LOCA - RHR line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line (RCIC connected) break	Fault Conditions	[VGL SFC 4-7.1]	The VGL components penetrating the primary containment form a barrier to confine radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1

SFC Table of CRD

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-7	Functions to plant instrument and control (except for safety protection function) (functions to control reactivity of the core in normal operational states)	-	No corresponding fault.	Normal Conditions	[CRD SFC 1-7.1]	The CRD through its FMCRD is part of the principal means (with the RC&IS) to deliver control rod insertion/withdrawal in plant normal operation conditions, the loss of which could lead to the demand of a Category A safety function (reactor scram).	B	3
2	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	-	No corresponding fault.	Normal Conditions	[CRD SFC 1-4.1]	The CRD through its FMCRD is the principal means to deliver maintenance of core sub-criticality in plant normal operation conditions.	A	1
3	1	Control of Reactivity	1-1	Functions to prevent excessive reactivity insertion	-	No corresponding fault.	Normal Conditions	[CRD SFC 1-1.1]	The CRD is the principal means to prevent excessive reactivity insertion caused by a CR drop event when the CR is separated from the ball nut.	A	1
4	2	Fuel Cooling	2-3	Function to make up reactor coolant with other system	-	No corresponding fault.	Normal Conditions	[CRD SFC 2-3.1]	The CRD is capable of supplying makeup coolant to the reactor to compensate for a small leakage and prevent it from resulting in a LOCA.	C	3
5	5	Others	5-20	Functions to maintain availability of CRs hydraulic insertion function and to recover CRs to normal unlatched state after rapid insertion	-	No corresponding fault.	Normal Conditions	[CRD SFC 5-20.1]	The CRD through its CRD Hydraulic System supports the delivery of reactor rapid shutdown through CRs hydraulic insertion by providing charging water to the HCUs..	C	3
6	5	Others	5-20	Functions to maintain availability of CRs hydraulic insertion function and to recover CRs to normal unlatched state after rapid insertion	-	No corresponding fault.	Normal Conditions	[CRD SFC 5-20.2]	Full insertion of the FMCRD with the electric motor after reactor was shut down by hydraulic insertion of the control rods provides recovery to unlatched state of the CRs.	C	3
7	4	Confinement/Containment of radioactive materials	4-4	Functions to contain radioactive material	-	No corresponding fault.	Normal Conditions	[CRD SFC 4-4.1]	The CRD piping and components contain radioactive material. A breach could lead to a release of radioactive material of dose consequences that are relatively low.	C	3
8	4	Confinement/Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault.	Normal and Fault Conditions	[CRD SFC 4-1.1]	The CRD portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
9	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 2.1 2.2 2.3 3.1 4.1 4.2 4.3 4.4 4.5 4.6 5.1 5.2 5.3 6.1 7.1 7.2 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3 10.4 11.2 11.3 11.8 11.9 11.10 11.11 11.12 13.1 17.1 17.2	Generator load rejection Partial loss of reactor flow (trip of 3 RIPs) Loss of reactor flow (trip of all RIPs) Feedwater controller failure - Maximum demand Recirculation flow control failure (runout of all RIPs) Loss of feedwater heating Reactor pressure regulator failure in the closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Control rod withdrawal error at start-up Control rod withdrawal error at power Control rod drop Inadvertent reactor scram (CRD pump trip) SRNM or APRM sensor failure Radiation monitor failure Short LOOP Medium LOOP Long LOOP Inadvertent opening of a SRV LOCA - RPV bottom drain line break LOCA - small line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line break LOCA - MS line break LOCA - RHR line break LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FWD line break LOCA outside PCV – small line break Inadvertent opening of all ADS Inadvertent start up of all injection system M/C power supply failure on electrical CCF D/C power supply failure on electrical CCF Loss of all RCW Loss of all RSW Loss of Class 1 HVAC SD - Control rod withdrawal error Internal fire in R/B Internal fire in Hx/B	Fault Conditions	[CRD SFC 1-3.1]	The CRD is the principal means to provide reactor rapid shutdown under RPS signal in conjunction with the CRs by performing CRs insertion, (actuation known as Scram), so that fuel design margins are not exceeded in the event of frequent faults and infrequent faults requiring reactor shutdown.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
					17.3 17.4 17.5 17.6 18.1 18.2 18.3	Internal fire in C/B Internal fire in MCR Internal missile in MCR Turbine missile Loss of UHS 10^{-3}/y earthquake DB earthquake					
10	1	Control of Reactivity	1-5	Functions of alternative reactivity control	1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 2.1 2.2 2.3 3.1 4.1 4.4 5.1 5.2 6.1 7.2 10.4 11.4 11.5 17.1 17.2 17.3 17.4 18.1 18.2	Generator load rejection Partial loss of reactor flow (trip of 3 RIPs) Loss of reactor flow (trip of all RIPs) Feedwater controller failure - Maximum demand Recirculation flow control failure (runout of all RIPs) Loss of feedwater heating Reactor pressure regulator failure in the closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Control rod withdrawal error at start-up Inadvertent reactor SCRAM (CRD pump trip) Short LOOP Medium LOOP Inadvertent opening of a SRV LOCA – small line break LOCA outside PCV – small line break ADS inadvertent opening due to SSLC failure MSIV inadvertent closure due to SSLC failure Internal fire in R/B Internal fire in Hx/B Internal fire in C/B Internal fire in MCR Loss of UHS 10^{-3}/y earthquake	Fault Conditions	[CRD SFC 1-5.1]	The CRD portions operated under the ARI signal are part of the secondary means to provide alternative reactor shutdown in the event of a frequent fault where reactor shutdown by Scram has failed (event known as Anticipated Transient Without Scram – ATWS).	A	2
11	1	Control of Reactivity	1-5	Functions of alternative reactivity control	-	No corresponding fault.	Fault Conditions	[CRD SFC 1-5.2]	The FMCRD Run-In mode of the FMCRD provides an alternative means of reactor shutdown by inserting the CRs with the FMCRD in the event reactor shutdown could not be achieved by RPS Scram, ARI and SLC.	C	3
12	1	Control of Reactivity	1-5	Functions of alternative reactivity control	-	No corresponding fault.	Fault Conditions	[CRD SFC 1-5.3]	The CRD in conjunction with the RC&IS is part of the means to deliver the SCRRI function which is an alternative means of controlling reactivity by inserting the CRs previously selected by controlling their FMCRD motor.	C	3

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
13	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	-	No corresponding fault.	Fault Conditions	[CRD SFC 1-4.2]	The CRD is the principal means to maintain the control rods inserted when shutdown by Scram in order to maintain the core sub-criticality.	A	1
14	1	Control of Reactivity	1-1	Functions to prevent excessive reactivity insertion	-	No corresponding fault.	Fault Conditions	[CRD SFC 1-1.2]	The CRD through its FMCRD is the principal means to prevent excessive reactivity insertion by prevention of control rod ejection.	A	1
15	1	Control of Reactivity	1-1	Functions to prevent excessive reactivity insertion	-	No corresponding fault.	Fault Conditions	[CRD SFC 1-1.3]	The CRD through its FMCRD is the principal means to prevent excessive reactivity insertion by limiting CRs drop speed.	A	1
16	4	Confinement/Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault.	Fault Conditions	[CRD SFC 4-7.1]	The CRD components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1

SFC Table of SLC

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-5	Functions of alternative reactivity control	1.1 1.2. 1.3 1.4 1.5 1.6. 1.7 1.8 2.1 2.2 2.3 3.1 4.1 4.2 4.4 5.1 5.2 6.1 7.2 10.4 17.1 17.2 17.3 17.4 18.1 18.2	Generator load rejection Partial loss of reactor coolant flow Partial loss of reactor flow (trip of 3 RIPs) FDW controller failure - Maximum demand Recirculation flow control failure (runout of all RIPs) Loss of FDW Reactor pressure regulator failure in closed direction Inadvertent control valve closure Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all FDW flow Control rod withdrawal error at start-up Control rod withdrawal error at power Inadvertent reactor SCRAM (CRD pump trip) Short term LOOP Medium term LOOP Inadvertent opening of a SRV LOCA – small line break LOCA outside PCV – small line break Internal fire in R/B Internal fire in Hx/B Internal fire in C/B Internal fire in MCR Loss of UHS 10^{-3}/y earthquake	Fault Conditions	[SLC SFC 1-5.1]	The SLC is the secondary means to provide reactor shutdown without CRs insertion, from full power operation, even during cycle equilibrium, to cold sub-critical condition by injecting the neutron absorbing solution into the reactor core in the event of ATWS design basis fault.	A	2
2	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	-	No claim for the design basis	Fault Conditions	[SLC SFC 1-4.1]	The SLC is the secondary means to maintain the reactor subcriticality without CRs insertion by injecting the neutron absorbing solution into the reactor core in the event of ATWS design basis fault.	A	2
3	4	Confinement/ Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No corresponding fault	Normal and Fault Conditions	[SLC SFC 4-1.1]	The SLC portions within the RCPB contain reactor coolant during normal conditions and form a pressure barrier during fault conditions the destruction of which would result in a loss of reactor coolant of radioactive consequences above the BSL.	A	1

	Top Claim for Mechanical System						Safety Functional Claim for the Mechanical System and Components (SFC)				
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)						
	PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		PCSR Ch.5 Section 6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule (Ref 12.1-1)		State	Claim ID	Claim Contents	Cat.	Class
4	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	[SLC SFC 4-7.1]	The SLC components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1

Appendix B: Safety Properties Claims Tables

The safety properties claims defined for mechanical systems are shown in the following table.

	SPC	Safety Properties Claims (SPC) Contents	SCDM SPC Guide word (Ref-12-1.5)
1	ME SPC1	Design provision against Single Failure Mechanical systems and their support systems are designed with redundancy against single failure of any dynamic component under the worst permissible system availability state so that single failure does not prevent the delivery of the corresponding safety functions.	• Fault Tolerance • Reliability
2	ME SPC2	Design provision against Common Cause Failure Mechanical systems are designed with independency between redundant components so that the failure of one dynamic component does not lead to a common cause failure that could prevent the delivery of the corresponding safety functions.	• Defence in Depth • Reliability
3	ME SPC3	Design provision against System Interfaces The mechanical interfaces between SSCs of different safety classes inside a mechanical system or between several systems are designed such that failure in a lower class item will not propagate to higher safety class items and jeopardise the delivery of the corresponding safety functions.	• Defence in Depth • Reliability
4	ME SPC4	Internal Hazards Protection Mechanical SSCs are protected or designed to withstand the effects of the following internal hazards so that they do not affect the delivery of the corresponding safety functions: (1) Internal flooding (2)Internal fire and explosion (3) Internal missiles (4)Dropped and collapsed loads (5) Pipe whip and jet impact (6)Internal blast (7) Electromagnetic Interference (EMI) (8)Miscellaneous hazards	• Fault Tolerance • Reliability
5	ME SPC5	External Hazards protection Mechanical SSCs are protected or designed to withstand the effects of the external hazards (Earthquakes, Loss of Offsite Power (LOOP)) so that they do not affect the delivery of the corresponding safety functions.	• Fault Tolerance • Reliability
6	ME SPC6	Actuation Provisions Mechanical systems are designed so that no human intervention is necessary for approximately 30 minutes following the start of the requirement for the safety function.	• Human Factors • Reliability
7	ME SPC7	Qualification Provision Mechanical SSCs are capable to deliver their safety functions under the associated operational and environmental conditions throughout their operational life.	• Qualification • Life Cycle • Reliability
8	ME SPC8	EMIT (Examination, Maintenance, Inspection and Test) Mechanical SSCs are designed with the capability for being tested, maintained and monitored during power operation and/or refuelling outages in order to ensure the capability to deliver the safety functions claimed without compromising their availability throughout their operational life.	• Life Cycle • Reliability • Layout and Accessibility • Radiation Protection
9	ME SPC9	Codes and Standards Mechanical components are designed manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected according to codes and standards commensurate to their Safety Class.	• Relevant Good Practice • Reliability

Note: the ME SPCs are derived based on the guide word approach as described in Chapter 5: General Design Aspects, Section 5.3: General Safety Design Bases.

The safety properties claims table of each system is provided in the following tables.

Ch.12.3 Reactor Coolant Systems and Associated Systems

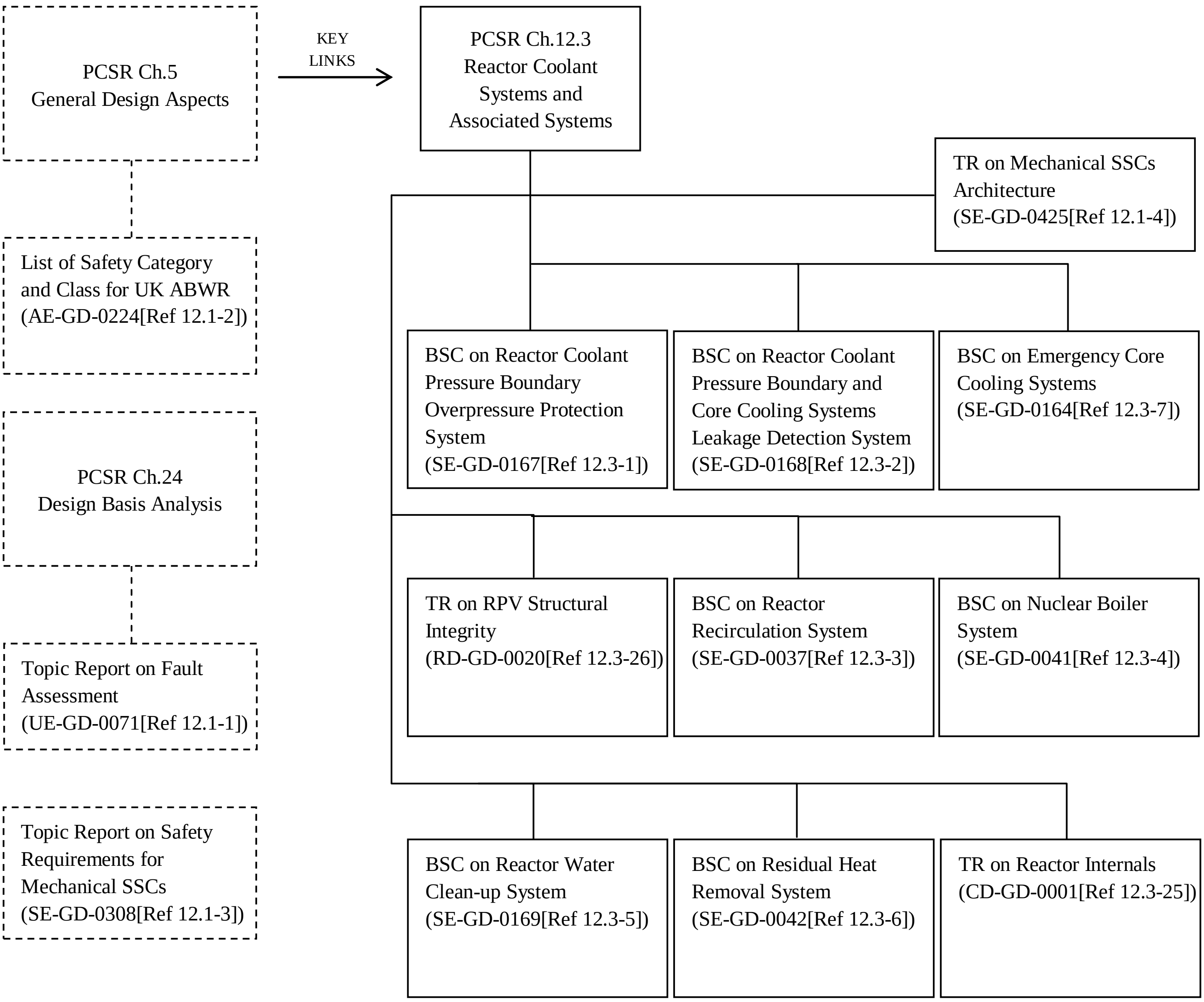
	Safety Cat. & Class	ME SPC1	ME SPC2	ME SPC3	ME SPC4	ME SPC5	ME SPC6	ME SPC7	ME SPC8	ME SPC9
		Design provision against Single Failure	Design provision against Common Cause Failure	Design provision against System Interfaces	Internal Hazards Protection	External Hazards protection	Actuation Provisions	Qualification Provision	EMIT	Codes and Standards
RCPB Overpressure Protection System	**A-1**	X	X	X	X	X	X	X	X	X
SRV	A-1	-	-	X	-	X	-	X	X	X
SRV Relief Accumulator	C-3	-	-	X	-	X	-	X	X	X
RCPB Leakage Detection System	A-1	X	X	X	X	X	X	X	X	X
Meters and transmitters for leakage detection and isolation function	A-1	-	-	X	-	X	-	X	X	X
Meters and transmitters for leakage detection and alarm function	C-3	-	-	X	-	X	-	X	X	X
RCPB	A-1	-	-	X	X	X	-	X	X	X
RRS	B-3	-	-	X	-	X	-	X	X	X
RIP	B-3	-	-	X	-	X	-	X	X	X
RMC components	B-3	-	-	X	-	X	-	X	X	X
NB	A-1	X	X	X	X	X	X	X	X	X
MS Flow Restrictor	A-1	-	-	X	X	X	-	X	X	X
MSIV	A-1	-	-	X	-	X	-	X	X	X
SRV	A-1	-	-	X	-	X	-	X	X	X
SRV Discharge Line Quencher	A-1	-	-	X	-	X	-	X	X	X
CUW	C-3	-	-	X	-	X	-	X	X	X
CUW pumps, heat exchangers and filter demineralizers portion for reactor coolant cleaning	C-3	-	-	X	-	X	-	X	X	X

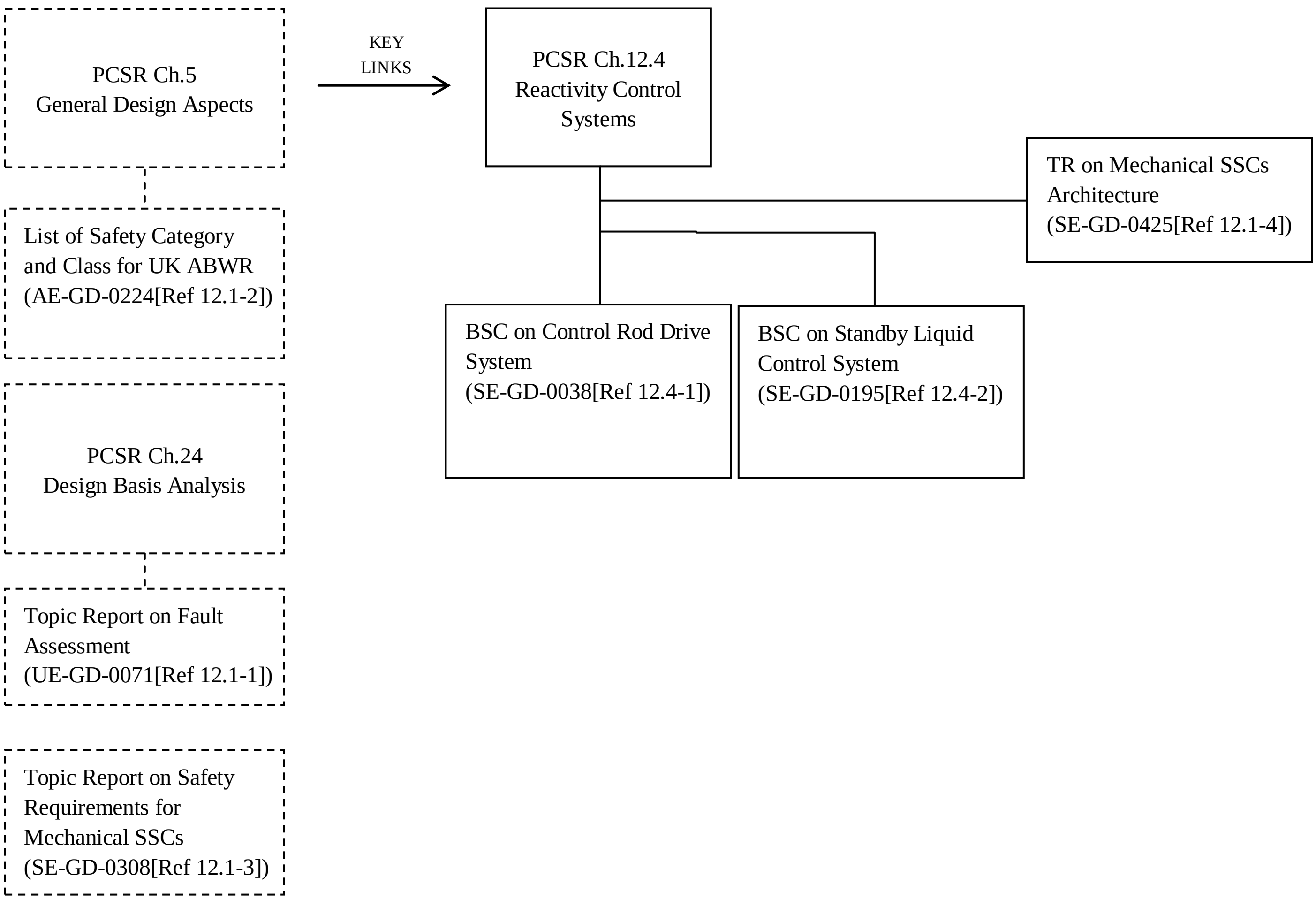
	Safety Cat. & Class	ME SPC1	ME SPC2	ME SPC3	ME SPC4	ME SPC5	ME SPC6	ME SPC7	ME SPC8	ME SPC9
		Design provision against Single Failure	Design provision against Common Cause Failure	Design provision against System Interfaces	Internal Hazards Protection	External Hazards protection	Actuation Provisions	Qualification Provision	EMIT	Codes and Standards
RHR	A-1	X	X	X	X	X	X	X	X	X
RHR SSCs	A-1	-	-	X	-	X	-	X	X	X
VGL	C-3	-	-	X	-	X	-	X	X	X
Piping and valves for leakage collection	C-3	-	-	X	-	X	-	X	X	X

Ch.12.4 Reactivity Control Systems

	Safety Cat. & Class	ME SPC1 Design provision against Single Failure	ME SPC2 Design provision against Common Cause Failure	ME SPC3 Design provision against System Interfaces	ME SPC4 Internal Hazards Protection	ME SPC5 External Hazards protection	ME SPC6 Actuation Provisions	ME SPC7 Qualification Provision	ME SPC8 EMIT	ME SPC9 Codes and Standards
CRD	**A-1**	X	X	X	X	X	X	X	X	X
Scram Portion	A-1	-	-	X	-	X	-	X	X	X
Hydraulic System Portion	C-3	-	-	X	-	X	-	X	X	X
SLC	**A-2**	X	X	X	X	X	X	X	X	X
SLC Boron injection portion	A-2	-	-	X	-	X	-	X	X	X
SLC Test portion	C-3	-	-	X	-	X	-	X	X	X

Appendix C: Document Map




PCSR Ch.5
General Design Aspects
KEY
LINKS
PCSR Ch.12.4
Reactivity Control
Systems
TR on Mechanical SSCs
Architecture
(SE-GD-0425[Ref 12.1-4])
List of Safety Category
and Class for UK ABWR
(AE-GD-0224[Ref 12.1-2])
BSC on Control Rod Drive
System
(SE-GD-0038[Ref 12.4-1])
BSC on Standby Liquid
Control System
(SE-GD-0195[Ref 12.4-2])
PCSR Ch.24
Design Basis Analysis
Topic Report on Fault
Assessment
(UE-GD-0071[Ref 12.1-1])
Topic Report on Safety
Requirements for
Mechanical SSCs
(SE-GD-0308[Ref 12.1-3])