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UK ABWR Generic Design Assessment

Generic PCSR Chapter 13 : Engineered Safety Features

Hitachi-GE Nuclear Energy, Ltd.

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Executive Summary

This mechanical systems chapter describes the safety case for the UK ABWR Engineered Safety Features (ESFs), consisting of the Containment Systems and the Emergency Core Cooling System (ECCS). It lists the high level Safety Functional Claims that are made on these, together with the Safety Property Claims (SPCs) that enable the process to demonstrate the compliance of these systems with the Nuclear Safety and Environmental Design Principles (NSEDPs).

The information provided includes: system design; functionality in normal operation and during faults; safety categorisation and classification; important support systems; safety case Assumptions, Limits and Conditions for Operation (LCO); resistance to hazards; and compliance with the ALARP principle.

The overall 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 PCSR 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 safety functional claims but do not specify requirements for design parameters on individual ESFs. 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 sub-systems and components within the ESFs are well advanced for GDA, being largely based on proven technology from the Japanese ABWR reference design. Additional risk reduction measures have been introduced (with reference to the J-ABWR design) in response to safety assessments undertaken in GDA. These include changes to the Reactor Core Isolation Cooling System (RCIC) pump design; redesign of the Flammability Control System (FCS) to replace active systems with Passive Autocatalytic Converters (PARs); and increase of the Residual Heat Removal System (RHR) heat removal capacity.

Other specific issues that have been considered in ALARP assessments include comparison of the methods of capturing or reducing fission products inside the primary containment with other reactor types, which concludes that the reference design is consistent with relevant good practice.
This chapter demonstrates that the risks associated with the design and operation of the ESFs for the UK ABWR are ALARP. It is acknowledged that further work will be required post GDA phase to develop the design and fully incorporate site specific aspects. This work will be the responsibility of any future licensee.

13.1 Introduction

This chapter of the PCSR describes the design features and various modes of operation of the ESFs for the UK ABWR. These consist of Containment Systems and Emergency Core Cooling Systems. This chapter lists all of the Safety Functional Claims (SFCs) that are made on these systems to maintain the High Level Safety Functions (HLSFs) under normal and fault conditions. It also lists all of the SPCs; these are used in lower tier reference documents to support the demonstration that the ESFs comply with the NSEDPs – [Ref 13.1-2]

13.1.1 Background

The ESFs of UK ABWR are designed to prevent or mitigate the impact of accidents and to contain potential radiological releases resulting from accidents. They are provided in the design to be additional to the normal means of providing the functions of core heat removal and containment of radioactive material that are described in other PCSR chapters.

The ESF 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 substantiate 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 Chapters 24 to 27 of the PCSR, demonstrate that the reactor and support systems are fault tolerant. The assumptions made in those analyses are fully consistent with the ESF design information and safety claims on the ESFs presented in this chapter.

13.1.2 Document Structure of Chapter 13

The following sections are included in this chapter:

<u>Section 13.2 Purpose and Scope:</u>
This section explains the objectives of the chapter, and lists the ESFs that are within the scope of this chapter.

<u>Section 13.3 Summary Description of Reactor Containment Systems and Associated Systems</u>:
The Reactor Containment Systems have the functions of preventing radioactive release to the environment in the event of a LOCA, or if complete prevention is not possible, limiting the amount of radioactive release to acceptable levels.
The scope of Structures, Systems and Components (SSCs) of the Reactor Containment Systems and associated systems that are described in this chapter is defined in sub-section 13.2.2 below.

<u>Section 13.4 Summary Description of Emergency Core Cooling Systems:</u>
The main safety functions of the ECCS are to mitigate the impact of a Loss of Coolant Accident (LOCA) by:

- limiting peak clad temperatures to prevent serious damage to the fuel in the core,
- suppressing the zirconium-water reaction of the fuel cladding as much as possible,
- providing long-term decay heat removal.

The scope of the ECCS SSCs described in this chapter is defined in sub-section 13.2.2 below.

Sections 13.3 and 13.4 both include the following for the relevant ESF SSCs:

- design information including system configurations;
- normal operation and safety functions delivered, and the various modes of operation;
- safety categorisation and classification of the SSCs;

- safety claims made on the SSCs, following the formal Claims-Arguments-Evidence (CAE) approach;
- the required support systems to fulfil the safety case claims.

Section 13.5 Assumptions, Limits and Conditions for Operation:
This section summarises the assumptions, limits and conditions for operation that are specified in greater detail in the Basis of Safety Case (BSC) documents for the SSCs in the scope of this Chapter.

Section 13.6 Summary of ALARP Justification:
This section provides a summary of the justification that the risks associated with the SSCs within Chapter 13 scope are acceptable (in terms of radiation dose consequences) and have been reduced to levels that are As Low as Reasonably Practicable (ALARP). This section refers to the results of the analysis in Chapters 24 to 27 that includes representation of systems within Chapter 13 scope.

Other relevant information is captured in Appendices as follows:

Appendix A – Safety Functional Claims Table for the Engineered Safety Features:
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 Topic Report (TR) of the related SSCs.

Appendix B – Safety Properties Claims Table for the Engineered Safety Features and Associated Systems:
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-13.1-3). 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 section 5.3 and the TR on Safety Requirements for Mechanical SSCs (Ref-13.1-7). Fulfilment of the requirements from the SPCs is justified in the BSC or TR of the related SSC as well as the TR on Mechanical SSCs Architecture (Ref-13.1-6).

Appendix C - Document Map for Level 2 documents that support this chapter:
This chapter is supported by a set of reference documents, primarily BSCs and their associated System Design Descriptions. Each BSC describes a specific system within the scope of Chapter 13, explaining where the arguments and evidence that substantiate the safety claims for those systems are presented. A full list of the Level 2 documents is provided within the document map in Appendix C.

This main links of this chapter with other GDA 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 Generic PCSR.

- The general principles for the identification of Assumptions, LCOs related to the systems within this chapter scope are described in Chapter 4 : Safety Management throughout Plant Lifecycle

- The categorisation of safety functions and safety classification of SSC 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 section related to EMIT of the BSC document of the systems within scope of this chapter.

- Hazard assessments (e.g. flooding, fire, rotating equipment related hazards, etc.) to demonstrate adequate performance of systems 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 reactor coolant systems which have functions linked to ESF such as the Nuclear Boiler System (NB) and the Residual Heat Removal System (RHR), etc. are described in Chapter 12 : Reactor Coolant Systems, Reactivity Control Systems and Associated Systems.

- The design of the systems 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 systems 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 systems scope of this chapter such as cooling water supply, Heating Ventilating and Air Conditioning (HVAC), compressed air supply, etc. , and the design of mechanical systems provided against beyond design basis faults and severe accidents is described in detail in Chapter 16 : Auxiliary Systems.

- Demonstration, using transient analysis, of the adequate performance of systems 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 systems within this chapter scope is in Chapter 25 : Probabilistic Safety Assessment.

- Substantiation of Human Based Safety Claims related to human interactions with systems 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.

13.2 Purpose and Scope

13.2.1 Purpose

The overall purpose of this chapter is to provide a definitive source of design information for all aspects of the ESFs that have an impact on nuclear safety under normal conditions or 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 GDA.

Specific objectives of the chapter are:

(1) Identify and describe the systems within the scope of the Reactor Containment Systems, ECCS, 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 systems 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 categorisation of those functions.

(3) Specify the safety classification of the SSCs within the scope of the chapter.

(4) Specify all relevant Safety Case Claims, and describe or provide pointers to where the detailed arguments and evidence for each claim can be found in the supporting BSCs, SDDs, and the detailed Level 3 design information, such as SDDs.
- These are Safety Functional Claims (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 13.1-3].

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

13.2.2 Scope

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

Reactor Containment Systems and associated systems –
- Primary Containment System
  - o Primary Containment Vessel (PCV)
  - o Primary Containment Isolation System (PCIS)
  - o Primary Containment Vessel Gas Control Systems, consisting of:
    - ▪ Atmospheric Control System (AC)
    - ▪ Flammability Control System (FCS)
  - o Containment Heat Removal Systems, consisting of:
    - ▪ Residual Heat Removal System (RHR)
    - ▪ Atmospheric Control System (AC)
    - ▪ Filtered Containment Venting System (FCVS)
  - o Drywell Cooling System (DWC)
- Secondary Containment System
  - o Reactor Building (R/B)
  - o Standby Gas Treatment System (SGTS)

Emergency Core Cooling System and alternative systems for core cooling –
- High Pressure Core Flooder System (HPCF)
- Reactor Core Isolation Cooling System (RCIC)
- NB Safety Relief Valves (SRVs) with safety and relief valve functions and the sub-set of 7 SRVs that constitutes the Automatic Depressurisation System (ADS) – the main source of design information for the SRVs is Chapter 12, but this chapter gives more details of operation of the SRVs as part of the ECCS.
- Low Pressure Flooder mode (LPFL) of the RHR – the RHR system configuration in LPFL mode is described in Chapter 12, but details of the LPFL mode of operation of the RHR as part of the ECCS is described in this chapter.
- Flooder System of Specific Safety Facility (FLSS) – the FLSS design is described in Chapter 16, but the mode of operation of the FLSS for injection into the reactor pressure vessel to cool the core during design basis faults as a backup of the ECCS is described in this chapter.
- Reactor Depressurisation Control Facility (RDCF) – the RDCF design is described in Chapter 16, but the mode of operation of the RDCF for depressurisation of the reactor to inject with the FLSS during design basis faults as a backup of the ECCS is described in this chapter.

Chapter 12 describes all modes of the RHR system, including LPFL mode which is an ESF, but further details of the containment heat removal and ECCS functions of the RHR are included in this chapter.

Chapter 16 describes all modes of the FLSS, RDCF and FCVS, which are systems that own functions during design basis faults, beyond design basis faults and severe accidents, but further details on their operation during design basis faults are provided in this chapter.

For links to GEP and CSA documentation, refer to Chapter 1. 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 Generic PCSR.

13.3 Containment System

13.3.1 Summary of Description

The reactor containment systems have the function of isolating the radioactive substances generated in the event of certain faults from the environment. A Loss of Coolant Accident (LOCA) is one of the most limiting of these faults.

By means of the reactor containment systems, leakage rates are kept below a specified low level, thus minimising the amount of potentially radioactive substances discharged into the atmosphere.

The containment systems consist of the following primary and secondary systems:

Containment systems:

(a) Primary Containment Vessel (PCV)
(b) Primary Containment Isolation System (PCIS)
(c) Primary Containment Vessel Gas Control System – consisting of the Flammability Control System (FCS) and the Atmospheric Control System (AC)
(d) Containment Heat Removal Systems – consisting of the RHR and the Containment Venting System (CVS)
(e) Drywell Cooling System (DWC)
(f) Reactor Building (R/B)
(g) Standby Gas Treatment System (SGTS)

The containment system could be sub-divided into Primary Containment (items (a) – (e)), and Secondary Containment (items (f) and (g)). Figure 13.3-1 shows boundary of primary and secondary containment.

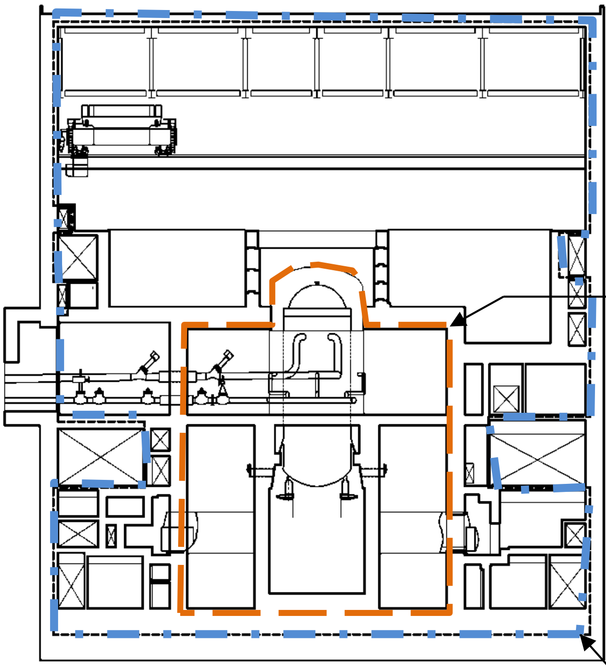

Figure 13.3-1: Primary and Secondary Containment Boundary

13.3.2 Design Intent

13.3.2.1 Design Requirement

Design requirements of the containment system can be classified into the following categories:

(1) Safety aspect requirements
(2) Structural integrity requirements

General design requirements of these groups are described below. The detailed design regarding structural integrity and civil engineering are detailed in Chapter 8 : Structural Integrity and Chapter 10 : Civil Works and Structures.

(1) Safety aspect requirements

(a) Confinement of radioactive substances
The basic function of the primary and secondary containment systems is to limit the leakage of radioactive substances [fission products (FP) and activation products] to the periphery of the nuclear power station (NPS) to minimise the dose equivalent for the general public around the premises.

(b) Air-tightness
The primary containment vessel is designed so that the air leakage rate from primary to secondary containment is 0.4% per day (%/d) or less of the containment volume, at normal temperature with 90 % of the maximum design pressure (related to [PCV SFC 4-7.3]).

The secondary containment facilities are designed so that the air in leakage rate into the secondary containment is 50% per day or less at a negative pressure of approximately 6 mm [water column]. This allows maintenance of the negative pressure to prevent radioactivity from leaking directly from the secondary containment into the atmosphere.

(c) Pressure Suppression
The primary containment vessel is designed so as to cool and condense the steam-water mixtures discharged into the drywell (D/W) using the Suppression Pool (S/P) water in the event of a LOCA, thus suppressing excessive increase of the pressure inside the containment vessel to maintain the containment vessel integrity (related to [PCV SFC 4-7.1]).

(d) Pressure and Heat Resistance
The primary containment vessel is designed so as to be able to withstand the maximum pressure and the maximum temperature produced as a result of any hypothetical LOCA, including an instantaneous complete break of one feedwater line or one main steam line, which are the limiting pipe break LOCA from a primary containment integrity perspective, as demonstrated in PCSR Chapter 24 ‘Design Basis Analysis’ (related to [PCV SFC 4-7.2]).

(e) Cooling Capability
The design makes it possible to cool the S/P water, contained in the Suppression Chamber (S/C) by means of the RHR System heat exchangers both during normal operation and during accidents, and also to cool the atmosphere inside the primary

containment vessel by spraying inside the containment vessel during accidents, thus maintaining the integrity of the containment vessel. (related to [RHR SFC 3-1.3], [RHR SFC 3-1.5] and [RHR SFC 4-7.1])

(f) Explosion-proof Capability
The necessary facilities are provided to maintain the hydrogen and oxygen concentration inside the primary containment vessel at levels below flammability limits for the conditions present following a LOCA, in order to eliminate the possibility of combustion of hydrogen gas (related to [FCS SFC 5-15.1]).

(g) Iodine Removal Capability
The necessary facilities are provided to remove iodine from inside the primary containment vessel in the event of an accident, and even if radioactive substances should leak into the secondary containment from the primary containment vessel, to purify the air inside the secondary containment and reduce the radioactive substances discharged into the atmosphere to an acceptable level by applying the Best Available Technique (BAT) (related to [SGTS SFC 4-7.2]).

(h) Isolation Valves
All pipework penetrating the primary containment vessel is provided with suitable isolation valves. The isolation valves shall be designed so as to prevent radioactive substances from leaking out from the containment vessel in the event of an accident.

(i) Protection against a Pipe Break
Facilities are designed to withstand the fluid jet forces associated with the flow from the postulated rupture of any pipe within the primary containment vessel. The arrangement of high energy pipework within the containment vessel is designed so that the containment vessel will not be damaged by whipping caused by jet reaction forces, and measures such as pipe restraints to prevent pipe whip are also incorporated as needed (related to [IH SFC 5-7.6]).

(j) Heat Removal and Overpressure Protection
The design makes it possible to prevent overpressure failure of the PCV and also to achieve the long-term PCV heat removal by means of venting through the Atmospheric Control System or the Filtered Containment Venting System (FCVS) in design basis faults (related to [AC SFC 3-2.1] and [FCVS SFC 3-2.1]).

(2) Structural integrity requirements

(a) Structural Strength
The containment vessel is designed so as to have a sufficient structural strength so that its integrity is not impaired by the static and dynamic loads anticipated during normal operation, abnormal operational transients and accidents including combination with seismic loads (related to [PCV SFC 4-7.4]).

(b) Prevention of Brittle Fracture
The steel components making up the containment vessel boundary are designed so that brittle fracture is prevented, taking the lowest design temperature into consideration (related to [PCV SFC 4-7.5]).

(c) Strength to Withstand Dynamic Loads

Within the primary containment vessel in the event of a LOCA, large amounts of coolant will flow out of the break and evaporate to steam as its pressure drops, and this is accompanied by movement of non-condensable gases from the D/W to the S/C. The steam which flows via the vents from the D/W into the S/C is condensed. In this process, the S/P water will move, producing various dynamic loads. When the safety/relief valves are actuated, dynamic loads are produced because the non-condensable gases and coolant in the discharge lines will flow into the S/C.

The primary containment vessel and the structures inside the containment vessel are designed to have sufficient structural strength so that their integrity is not impaired when subjected to these dynamic loads (related to [PCV SFC 4-7.6]).

13.3.3 Primary Containment Facility System Design Description

13.3.3.1 Primary Containment Vessel

The Primary Containment Vessel (PCV) is a reinforced-concrete structure with an internal steel liner. Figure 13.3-2 shows a typical cross section of the ABWR PCV. It consists of components such as a cylindrical D/W surrounding the Reactor Pressure Vessel (RPV), a cylindrical S/C and a base mat. Inside the containment vessel, there are the reinforced-concrete diaphragm floor, which partitions the D/W from the S/C, and the steel RPV foundation (hereinafter in this chapter called the “pedestal”). The steel vent pipes which connect the D/W and the S/C are built into the pedestal. Furthermore, the PCV is equipped with vacuum breakers, containment vessel penetrations and isolation valves.

In the event of a LOCA, the steam-water mixture released into the D/W is led into the S/P water through the vent pipes. The steam is cooled and condensed by this pool water, thus suppressing the pressure rise in the D/W. Any radioactive substances are retained inside the containment vessel.

The temperature inside the D/W during normal operation is kept within a constant temperature range by the Drywell Cooling System (DWC).

The outlines of primary containment vessel and main structures or components in the PCV are as follows. Major specified values are shown in Table 13.3-1. The detailed design of the PCV and the arrangement inside is addressed in Chapter 10 : Civil Works and Structures.

(1) Primary Containment Vessel

The PCV contains the D/W and the S/C in which water is stored. The D/W and the S/C are reinforced-concrete pressure vessels lined on the inside with steel liners for preventing leakage. The top head of the upper D/W and the containment vessel penetrations are made of steel.

(2) Diaphragm Floor and Vent Pipes

The diaphragm floor is provided in order to separate the containment vessel into the D/W and the S/C.

The vent pipes are provided in order to transmit the steam which is released into the D/W in the event of a LOCA from the D/W into the pool water in the S/C, so that the steam is completely condensed there.

(3) Vacuum Breakers

Vacuum breakers are provided to automatically relieve any pressure difference if condensation of steam inside the D/W should proceed after a LOCA to the point where the D/W pressure drops below the S/C pressure. In such cases, the vacuum breakers prevent backflow of the pool water from the S/C into the D/W, as well as avoiding damage of the diaphragm floor and pedestal by excessive differential pressure between the D/W and the S/C.

(4) Containment Vessel Penetrations

(a) Pipe and electrical cable penetrations
Pipe and electrical cable penetrations are contained within welded steel enclosures which are seal-welded directly to the PCV steel lining plates of the reinforced-concrete containment vessel via steel reinforcement plates.

(b) Personnel air locks and equipment hatches
The personnel air locks consist of double doors provided with interlocks so that both of these doors will not open simultaneously. The equipment hatches into the D/W, the D/W head and the S/C entrance are sealed by double gaskets.

Design Bases

This section describes the design bases for the PCV.
The PCV has been designed to meet the following SFCs. The linkage between the SFCs of the PCV with the FSFs and the HLSFs is shown in the Appendix A. The FSFs and the HLSFs are defined in Chapter 5 : General Design Aspects.

Normal Conditions

(1) The air leakage ratio of the RCCV is 0.4 percent per day or less of free volume of the containment at ordinary temperature and with a 90 percent of the maximum design pressure [PCV SFC 4-7.3] ([Ref-13.3-18]).

(2) The RCCV and structures within the RCCV have a structural strength that maintains integrity when assumed static load and dynamic load generated in normal condition (and fault conditions) are appropriately combined with the relevant seismic load [PCV SFC 4-7.4] ([Ref-13.3-18]).

(3) As for the steel parts in the RCCV, brittle fracture is prevented by taking the lowest design temperature (10°C) into consideration [PCV SFC 4-7.5] ([Ref-13.3-18])

(4) The RCCV has a structural strength that maintains integrity when assumed static load and dynamic load generated in normal condition (and in fault conditions) are appropriately combined with the relevant seismic load to support SSCs within RCCV (and/or to support the RCCV) [PCV SFC 5-17.1] ([Ref-13.3-18]).

Fault Conditions

(1) Any steam released into the RCCV from a possible pipe rupture in the primary system will be condensed by the S/P, and any significant pressure rise will be suppressed [PCV SFC 4-7.1] ([Ref-13.3-18]).

(2) The RCCV can withstand the maximum excessive pressure and temperature caused by the defined LOCA events including piping break such as instantaneous, complete and double-ended guillotine break of one feedwater piping or one main steam piping [PCV SFC 4-7.2] ([Ref-13.3-18]).

(3) The air leakage ratio of the RCCV is based on 0.4 percent per day or less of free volume of the containment at ordinary temperature and with a 90 percent of the maximum design pressure [PCV SFC 4-7.3] ([Ref-13.3-18]).

(4) The RCCV and structures within the RCCV have a structural strength that maintains integrity when assumed static load and dynamic load generated in fault conditions (and normal condition) are appropriately combined with the relevant seismic load [PCV SFC 4-7.4] ([Ref-13.3-18]).

(5) As for the steel parts in the RCCV, brittle fracture is prevented by taking the lowest design temperature (10°C) into consideration [PCV SFC 4-7.5] ([Ref-13.3-18]).

(6) The RCCV and structures within the RCCV have sufficient structural strength to maintain integrity against the following hydrodynamic loads [PCV SFC 4-7.6] ([Ref-13.3-18]).
- Gas / steam release
- Pool swell
- Steam condensation (oscillation / chugging loads)
- Annulus Pressurisation

(7) The RCCV has a structural strength that maintains integrity when assumed static load and dynamic load generated in fault conditions (and in normal condition) are appropriately combined with the relevant seismic load to support SSCs within RCCV (and/or to support the RCCV) [PCV SFC 5-17.1] ([Ref-13.3-18]).

(8) The RCCV and structures within the RCCV have sufficient structural strength to maintain integrity against the following hydrodynamic loads to support SSCs (and/or to support the RCCV) [PCV SFC 5-17.2] ([Ref-13.3-18]).
- Gas / steam release
- Pool swell
- Steam condensation (oscillation / chugging loads)
- Annulus Pressurisation

Assumptions, Limits and Conditions for Operations

In order to ensure that the Primary Containment Vessel is operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCO, 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 BSC on Reinforced Concrete Containment Vessel [Ref-13.3-18].

- Primary Containment, Two Primary Containment Air Locks and Eight Wetwell-to Drywell Vacuum Breakers shall be operable during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.
- S/P water level shall be within prescribed limit during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.

Table 13.3-1: Pressure and Temperature of Primary Containment Vessel

	Item	Specified value: D/W	Specified value: S/C
Specified value	Pressure in normal operation (inner pressure)	5 kPa [gauge] (average)	5 kPa [gauge] (average)
	Maximum design pressure (inner pressure)	310 kPa [gauge]	310 kPa [gauge]
	Maximum design pressure (outer pressure)	14 kPa [gauge]	14 kPa [gauge]
	Temperature in normal operation	57 °C (average)	35 °C (maximum)
	Maximum design temperature	171 °C	104 °C
	Lowest design temperature*1	10 °C	10 °C
	Leakage rate*2	0.4 %/d max.	0.4 %/d max.
	Diaphragm floor design pressure difference	173 kPa [diff]	
	Vent pipe Maximum design pressure (inner pressure)	173 kPa [diff]	
	Maximum design temperature	171 °C	

*1: In normal operation.
*2: Leakage rate at 0.9 times the maximum design pressure of containment vessel is 0.4 %/d or lower.

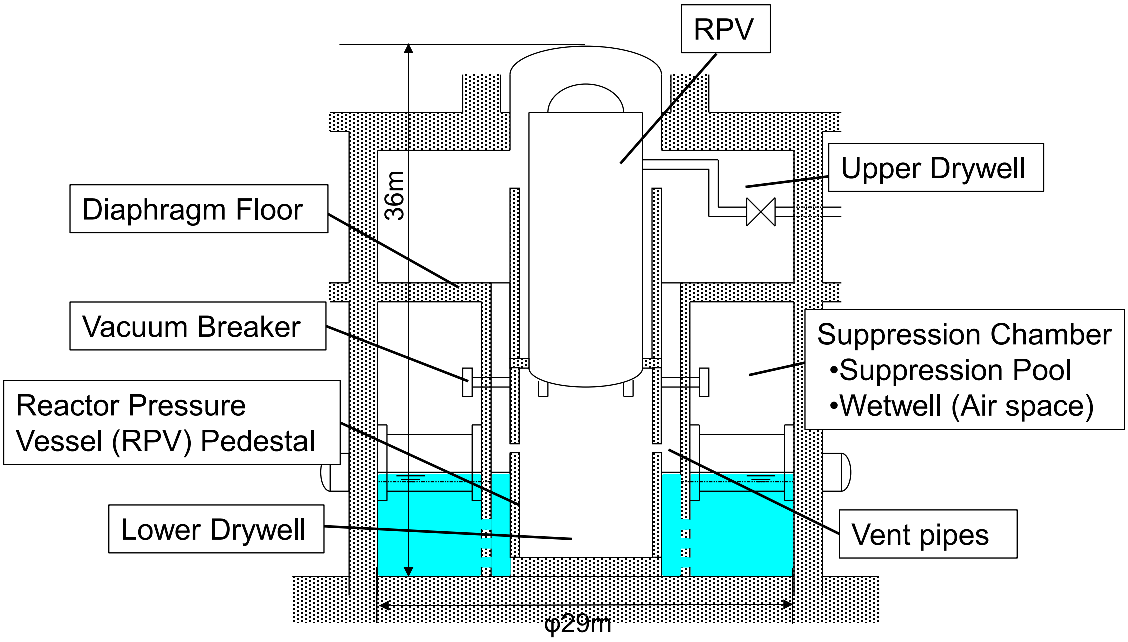

Figure 13.3-2: Cross Section of the ABWR PCV

13.3.3.2 Primary Containment Isolation System

System Summary Description

This section is a general introduction to the Primary Containment Isolation System (PCIS), in which the system roles, system functions and basic configuration are briefly described. The safety case of the PCIS is justified in the Containment Isolation System BSC [Ref-13.3-1] and description of the system is also included.

(1) System Roles

The main role of the PCIS is to provide protection against the release of radioactive material from the Primary Containment Vessel (PCV) to the environment in the case of faults that lead to the release of reactor core coolant inside or outside the PCV.

(2) Functions Delivered

The PCIS provides protection against the release of radioactive material from the PCV to the environment during faults above stated by isolating the pipes of the systems penetrating the primary containment and thus forming a barrier to confine the radioactive material. This barrier is formed by the piping, the isolation devices and the containment itself is called the Primary Containment Vessel Boundary (PCVB).

(3) Basic Configuration

The PCIS consists of the isolation devices (valves) and controls required for the isolation of the piping penetrating the containment in support of confinement function.

(4) Design Bases

This section describes the design bases for the PCIS.

The PCIS has been designed to meet the following Safety Functional Claim (SFC). The linkage between the SFCs of the PCIS with the Fundamental Safety Functions (FSFs) and the HLSFs is shown in the Appendix A. The FSFs and HLSFs are defined in Chapter 5 : General Design Aspects.

The PCIS provides protection against the release of radioactive material from the PCV to the environment during fault conditions by isolating the pipes of the systems penetrating the primary containment vessel and thus forming a barrier to confine the radioactive material within the PCVB.

This leads to the following SFCs, which apply to the systems forming part of the PCVB. (Refer to the Chapter 1 : Introduction for acronyms and abbreviations used for the systems that form the PCVB barrier).

(a) The RRS 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. [RRS SFC 4-7.1]

(b) The NB 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. [NB SFC 4-7.3]

(c) 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]

(d) 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]

(e) The RCIC 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. [RCIC SFC 4-7.1]

(f) The HPCF 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. [HPCF SFC 4-7.1]

(g) 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]

(h) 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]

(i) The AC 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. [AC SFC 4-7.1]

(j) The FCVS 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. [FCVS SFC 4-7.1]

(k) The RCW 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. [RCW SFC 4-7.1]

(l) The IA 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. [IA SFC 4-7.1]

(m) The SA 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. [SA SFC 4-7.1]

(n) The HPIN 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. [HPIN SFC 4-7.1]

(o) The RD 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. [RD SFC 4-7.1]

(p) The SAM 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. [SAM SFC 4-7.1]

(q) The SPCU 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. [SPCU SFC 4-7.1]

(r) The RDCF 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. [RDCF SFC 4-7.1]

(s) The PRM mechanical 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. [PRM SFC 4-7.1]

(t) The HNCW 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. [HNCW SFC 4-7.1]

(u) The MUWC 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. [MUWC SFC 4-7.1]

(v) The FLSS 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. [FLSS SFC 4-7.1]

(w) The FLSR 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. [FLSR SFC 4-7.1]

(x) The RVI mechanical 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. [RVI SFC 4-7.1]

(y) The PCV leak test facility 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. [PCV L/T SFC 4-7.1]

(z) The CAMS mechanical 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. [CAMS SFC 4-7.1]

(aa) The TIP mechanical 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. [TIP SFC 4-7.1]

(bb) The MUWP 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. [MUWP SFC 4-7.1] (cc) The VGL 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. [VGL SFC 4-7.1]

(dd) The ANI 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. [ANI SFC 4-7.1]

These safety functions are categorised as Safety Category A and the components to deliver them are designed to meet Safety Class 1 requirements.

System Design Description

This section describes the design of the PCIS to support and justify the delivery of all the SFCs. Additional design description can be found in the BSCs on PCIS [Ref-13.3-1].

(1) System Design and Operation

All the piping penetrating the PCV is grouped and configured as described below depending on the original configuration and function of the piping. For further details refer to the “Primary Containment Isolation System Design Philosophy” [Ref-13.3-19].

Gr. A: Piping that forms part of the Reactor Coolant Pressure Boundary (RCPB Piping)

1) Gr. A.1: Systems not delivering other safety mitigation functions different from isolation in the event of LOCA

- If the pipe penetrating the PCV forms an open piping system, it is provided with two isolation valves, one on each side of the PCV. These valves are designed to automatically close under LOCA conditions in order to deliver containment isolation function. See Figure 13.3-3.

2) Gr. A.2: Systems delivering other safety functions for LOCA mitigation in addition to isolation function

- If the pipe penetrating the PCV forms an open piping system, it is provided with two isolation valves, one on each side of the PCV. However, the valves are designed not to automatically close under LOCA conditions in order to prioritise the delivery of the other safety mitigation functions. See Figure 13.3-3.
- If the pipe penetrating the PCV forms a closed piping system, the pipe is regarded as an integral part of the containment and rated against LOCA conditions. Therefore, there is no requirement to provide any valves to isolate the pipe. However, one single A-1 valve is always provided for defence-in-depth. See Figure 13.3-4.

Gr. B: Piping that does not form part of the RCPB (Non-RCPB Piping)

1) Gr. B.1: Systems delivering other safety functions for LOCA mitigation in addition to isolation function

- If the pipe penetrating the PCV forms an open piping system, it is provided with two isolation valves without requirement to provide physical separation by the PCV. The valves are designed not to automatically close under LOCA conditions in order to prioritise the delivery of the other safety mitigation functions. See Figure 13.3-5.
- If the pipe penetrating the PCV forms a closed piping system, the pipe is regarded as an integral part of the containment and rated against LOCA conditions. Therefore, there is no requirement to provide any valves to isolate the pipe. However, one single A-1 valve is always provided for defence-in-depth and is designed not to automatically close under LOCA conditions in order to prioritise the delivery of the other safety mitigation functions. See Figure 13.3-6.

2) Gr. B.2: Systems not delivering other safety mitigation functions different from isolation in the event of LOCA

- If the pipe penetrating the PCV forms an open piping system, it is provided with two isolation valves without requirement to provide physical separation by the PCV. These valves are designed to automatically close under LOCA conditions in order to deliver containment isolation function. See Figure 13.3-5.
- If the pipe penetrating the PCV forms a closed piping system, the pipe is regarded as an integral part of the containment and rated against LOCA conditions. Therefore, there is no requirement to provide any valves to isolate the pipe. However, one single A-1 valve is always provided for defence-in-depth. See Figure 13.3-6.

Based on these design principles, all the pipes penetrating the PCV are provided with the corresponding isolation means as detailed in the "Basis of Safety Cases on Containment Isolation System" [Ref-13.3-1].

(2) Main Support Systems

(a) Instrumentation and Control Systems

Instrumentation and control are provided to measure and monitor the operating conditions of the PCIS components necessary for the delivery of primary containment isolation. The relevant instrumentation and control provisions are described as follows:

(i) Automatic closure of the required isolation valves is controlled by the logic of the Safety System Logic and Control System (SSLC), which governs the PCIS that gives the order to close the valves.

(ii) All isolation valves that are part of systems not required for the delivery of other safety functions different from isolation in the event of faults are automatically closed under the isolation signals if they are not already passively isolated (check valves or locked-close valves in plant normal operation conditions).

(iii) The different systems that form part of the LDS monitor these process parameters in order to detect leakage of reactor coolant and transmit some of the isolation signals to the corresponding isolation valves. These valves might receive other isolation signals, but they are not described here because they are not related to the delivery of HLSF 4-7.

(iv) Valves which are required automatic isolation, are also provided with a remote manual switch for actuation from MCR HMI as defence-in-depth. Furthermore, the design ensures that resetting the isolation signal does not result in the automatic reopening of the containment isolation valves.

(v) Isolation valves that are part of a system required to deliver other safety functions different from isolation during faults are designed not to automatically close by the PCIS isolation signals because the priority is not the confinement function. Nonetheless, they are provided with a remote manual switch controlled by the SSLC to be operable from the MCR as defence-in-depth.

(vi) Power-operated (motor-operated or pneumatically-operated) containment isolation valves have indicating switches in the MCR HMI to show the valve status. Loss of power to the motor-operated valves is detected and annunciated in the MCR by an alarm.

(vii) Provisions for administrative controls and/or locks ensure that the position of all manual isolation valves is correctly maintained and known.

(b) Power Supply System

(i) Electrical power to the isolation valves is supplied by functionally independent and separated divisions of the Safety Class 1 AC Power Distribution System and the Safety Class 1 DC Power Distribution System depending on the type of valve.

(ii) Valves required to automatically close by the isolation signals in the event of faults are connected to the emergency diesel generators to supply them with power in the event of Loss of Offsite Power (LOOP).

(3) Architecture

(a) Redundancy

N+2 redundancy is never applied because planned maintenance of these valves does not take place during plant operation. During maintenance for plant outage, the piping and the valves are isolated before entering maintenance activities, and therefore the isolation function is ensured beforehand.

N+1 redundancy is applied for all ‘open piping systems’ instead, i.e. those piping that do not form a closed system. All open piping is therefore provided with two redundant isolation valves against single failure (this is also applicable to their support systems).

N redundancy is required for ‘closed piping systems’, i.e. those piping that take from and return to PCV or piping that forms a ‘dead end’. There is no requirement to provide any valves for these closed systems, as all piping is considered an integral part of the containment and rated against LOCA conditions where required. However, at least one single A-1 valve is always provided for defence-in-depth. Additionally, closed piping that belongs to the RCPB is provided with one check valve inside the PCV from the point of view of the delivery of HLSF 4-1 (Reactor Coolant Pressure Boundary), which accounts for N+1 redundancy.

(b) Independence

As a general rule where there is requirement for N+1 redundancy, the valves and their respective actuators are arranged with functional independence and physical separation such that failure of any dynamic component does not lead to a common cause failure that could prevent the delivery of the isolation function. Furthermore, support systems for actuation, such as power supply, come from independent electrical divisions.

Assumptions, Limits and Conditions for Operations

In order to ensure that the PCIS is operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCO, 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 Primary Containment Isolation System [Ref-13.3-1].

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

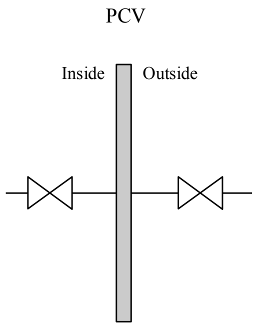

Figure 13.3-3: RCPB Piping Pattern 1 (Gr. A1 and A2)

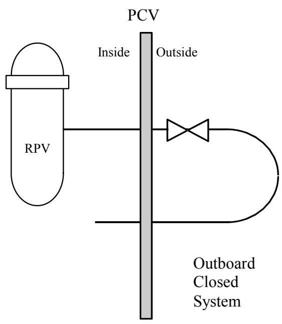

Figure 13.3-4: RCPB Piping Pattern 2 (Gr. A2)

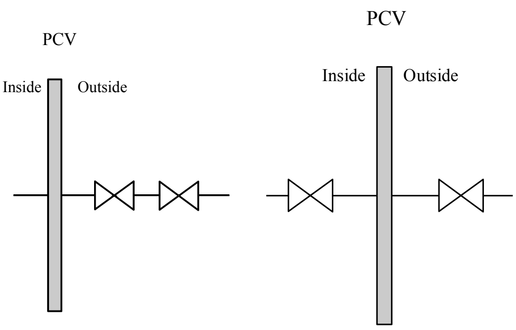

Figure 13.3-5: Non-RCPB Piping Patterns 1, 2 (Gr. B1 and B2)

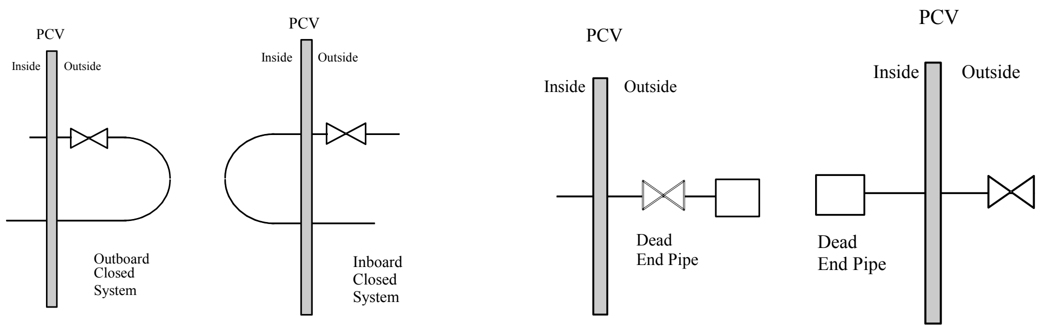

Figure 13.3-6: Non-RCPB Piping Patterns 3, 4, 5, 6 (Gr. B1 and B2)

13.3.3.3 Primary Containment Vessel Gas Control Systems

The PCV Gas Control Systems include the Flammability Control System (FCS) and the Atmospheric Control System (AC) with the principal role of maintaining an inert and non-explosive atmosphere within the PCV. The systems are designed to prevent build-up of hydrogen and oxygen which could be generated within the reactor and released into the PCV in a design basis event.

The FCS is provided to control the potential build-up of hydrogen from radiolysis of water.

The AC is provided to establish and maintain an inert atmosphere within the PCV during all plant operating modes except during shutdown for refuelling outages or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power.

Flammability Control System (FCS)
System Summary Description

This section is a general introduction to the FCS, in which the system roles, system functions, system configuration and modes of operation are briefly described. The FCS safety case is justified in the BSC on PCV Gas Control Systems [Ref-13.3-2].

(1) System Roles

The main role of the FCS is to control the potential build-up of hydrogen from design basis metal-water reaction, which might occur after a LOCA, and from the radiolysis of water. The objective is to limit the concentration of flammable gases (hydrogen and oxygen) below the flammability limits. This prevents the possibility of a reaction of these flammable gases in the PCV which would potentially result in an excessive increase in pressure and temperature due to the heat released by the reaction.

(2) Functions Delivered

The FCS limits the hydrogen and oxygen concentrations inside the PCV below 4 percent (volumic) and 5 percent (volumic) respectively.

(3) Basic Configuration

The FCS consists of Passive Auto-catalytic Recombiners (PARs). The PAR consists of a catalyst and a housing. The main components are summarised as follows:

(a) PARs: 5 units (4 units are located in the D/W, 1 unit is located in the W/W) *

* For the purposes of the assessments performed in GDA, the FCS has been assumed to contain five units of PARs. It has also been assumed 4 units are set in the D/W and 1 unit is set in the W/W

(4) System Design Bases

This section describes the design bases for the FCS.

The FCS has been designed to meet the following SFC. The linkage between the SFCs of the FCS with the FSFs and the HLSFs is shown in Appendix A. The FSFs and HLSFs are defined in Chapter 5 : General Design Aspects.

The FCS backs up the confinement function by maintaining the hydrogen and oxygen concentrations in the PCV below the flammability limits through recombination of the

gases generated and accumulated in the PCV that might occur after design basis faults such as LOCA. [FCS SFC 5-15.1]
This function is categorised as Safety Category B and the components necessary to deliver it are classified as Safety Class 2 safety components.

System Design

This section describes the design of the FCS to satisfy the design bases. The FCS safety case is justified in the BSC on PCV Gas Control System [Ref-13.3-2].

(1) Overall Design and Operation

The PARs passively initiate hydrogen and oxygen recombination when the concentration of hydrogen and oxygen in the PCV increases.
Gas with hydrogen content comes into contact with the catalyst which initiates a chemical reaction between hydrogen and oxygen to form water vapour. The chemical energy released by this exothermal reaction heats up the produced gas. Then, the gas ascends through the upper opening of the PAR due to buoyancy, which makes room for new hydrogen-rich gas coming from below PARs to react at the catalyst surface.

(2) Equipment Design and Operation

(a) FCS Recombiner (PARs)
(i) Configuration
The FCS is provided with five sets of PARs (*) so that the hydrogen and oxygen concentrations in the gases inside the PCV do not exceed 4 percent and 5 percent (volumic) respectively after a LOCA, assuming the atmosphere within the containment is inert prior to the LOCA. Four of the PARs are set in the D/W and one is set in the W/W (*). Each PAR consists of a vertical metal channel (housing), with openings at its lower and upper end, a hood over the upper opening and catalyst cartridges.
* For the purposes of the assessments performed in GDA, the FCS has been assumed to contain five units of PARs. It has also been assumed 4 units are set in the D/W and 1 unit is set in the W/W.

(ii) Performance
The PAR is designed to perform as follows to ensure the delivery of [FCS SFC 5-15.1].

Table 13.3-2: PAR Capacity

Basic Specifications	
Depletion Rate	More than 0.5 kg/h at 4.0 vol% hydrogen and rich oxygen (more than 4.0 vol%) **

** For the purposes of the assessments performed in GDA, depletion rate has been assumed to be more than 0.5 kg/h at 4.0 vol%.

(3) System Architecture

(a) Redundancy
The FCS consists of four PARs in the D/W such that, single failure of a PAR in the D/W due to a jet load caused by a piping break does not prevent the delivery of the safety function.

(b) Independence
The components of the PARs are independent and physically separated to prevent failure of a PAR from leading to a common cause failure of the other PARs.

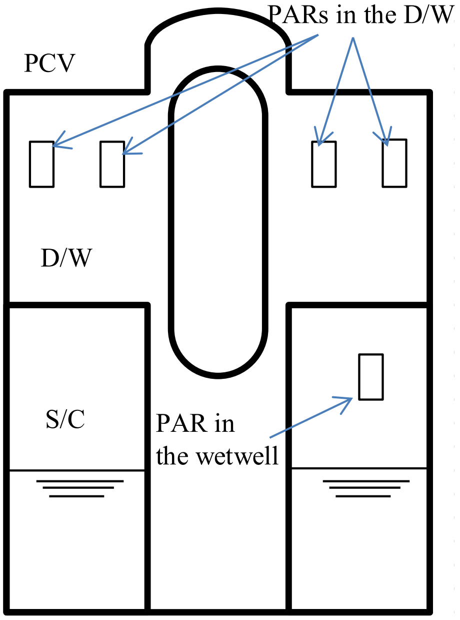

Figure 13.3-7: Outline of the Flammability Control System

Atmospheric Control System (AC)

Summary Description

This section is a general introduction to the AC where the system roles, system functions, system configuration and modes of operation are briefly described. The AC safety case is justified in the BSC on PCV Gas Control Systems [Ref-13.3-2]. The AC is described in detail in the system specifications [Ref-13.3-5] and P&IDs [Ref-13.3-6] and [Ref-13.3-7].

(1) System Roles

The main role of the AC is to inert the atmosphere within the PCV by replacing the air with nitrogen in order to maintain the concentration of oxygen in the containment at a sufficiently low level during normal operation to prevent the combustion of hydrogen generated in fault condition.

(2) Functions Delivered

The AC is designed to perform the following functions:

(a) Inerting (plant start-up)
The AC establishes an inert atmosphere throughout the PCV following a refuelling outage, or other situations when the PCV has been purged with air, by introduction of nitrogen to reduce oxygen concentration to no greater than 4.0vol%.

(b) PCV Pressure Control (power operation)
During power operation of the plant, the AC maintains a slightly positive pressure inside the PCV to prevent leakage of outside air into the inert PCV atmosphere. By supplying nitrogen gas to the D/W or the S/C, the AC compensates the normally occurring leakage to the outside. By venting the inside atmosphere of the PCV, the AC prevents excessive pressure increases. When venting, the gases inside the PCV can be exhausted through two paths:

(i) Through the Heating, Ventilation and Air Conditioning system (HVAC) when radioactivity of the exhaust gases is low
(ii) Through the SGTS when an increase in exhaust gases radioactivity has been detected.

(c) De-Inerting (shutdown and refuelling outages)
At the time of plant shutdown, the AC replaces the nitrogen atmosphere in the PCV with air supplied by the Reactor Building (R/B) HVAC to allow personnel access.

(d) Containment heat removal and overpressure protection (fault scenarios)
The AC provides PCV heat removal and prevents PCV overpressurisation under design basis faults by venting the PCV gases to the main stack through the hardened vent line bypassing the SGTS filter trains.

(e) Nitrogen Supply to Pneumatic Equipment inside the PCV
The makeup portion of AC supplies nitrogen gas to the High Pressure Nitrogen Gas Supply System (HPIN), and to Instrument Air System (IA) via HPIN to charge the Safety Relief Valve (SRV) accumulators and provide nitrogen gas to the instrumentation and the pneumatic equipment inside the PCV as well as the equipment requiring nitrogen gas in the R/B.

(f) Pressurisation of the PCV for Tests
The AC supplies nitrogen gas to support the initial PCV structural integrity test and the periodic integrated leak rate test of the PCV.

(3) Basic Configuration

The main components forming the AC are summarised as follows:

(a) AC Nitrogen Gas Supply Machine:
  (i) AC Liquid Nitrogen Evaporator 1 unit
  (ii) AC Nitrogen Gas Supply Machine 1 unit
(b) AC Nitrogen Gas Heater 1 set
(c) Piping and valves
(d) Instruments and Controllers

Figure 13.3-8 shows an outline of the AC basic configuration.

(e) PCV Instrumentation
The AC includes the following PCV instrumentation:
  (i) Temperature of the D/W and the S/C
  (ii) Pressure in the D/W and the S/C
  (iii) Water Level of the S/P

(4) Design Bases

This section describes the design bases for the AC.

The AC has been designed to meet the following SFCs. The linkage between the SFCs of the AC with the FSFs and the HLSFs is shown in the Appendix A. The FSFs and HLSFs are defined in Chapter 5 : General Design Aspects.

<u>Normal Conditions</u>

(a) The AC supplies nitrogen gas to maintain inert condition within the PCV during inerting operation and plant power operation, which prevents hydrogen combustion in case of design basis faults. [AC SFC 4-17.1]

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

(b) The AC supplies nitrogen gas to maintain slight positive pressure within the PCV during plat power operation, which prevents air in-leak from secondary containment to the PCV. [AC SFC 4-17.2]

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

(c) The AC supports the pneumatic equipment inside the PCV requiring nitrogen gas supply such as the SRV accumulators, the inboard MSIV accumulators, instrumentation, etc., and equipment using nitrogen gas inside the R/B to deliver their respective safety functions by supplying nitrogen gas through the HPIN. [AC SFC 5-13.1]

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

Fault Conditions

(d) The AC is a secondary means to deliver long term PCV heat removal and overpressure protection in the event of frequent faults where the primary long-term containment heat removal means (RHR) has failed. [AC SFC 3-2.1]

This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements. It is developed and justified in section 13.3.3.4 of this chapter related to the Containment Heat Removal Systems.

(e) The AC 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. [AC SFC 4-7.1]

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 in section 13.3.3.2 of this chapter related to the PCIS.

System Design Description

This section describes the design of the AC to support and justify the delivery of [AC SFC 4-17.1], [AC SFC 4-17.2] and [AC SFC 5-13.1]. Additional design description can be found in [Ref-13.3-2], [Ref-13.3-5], [Ref-13.3-6] and [Ref-13.3-7].

(1) System Design and Operation

(a) Inerting Mode
The AC supplies nitrogen gas and exhausts the air from the PCV simultaneously, to provide an inert atmosphere in the PCV. In the inerting operation, nitrogen gas is supplied from the AC Liquid Nitrogen Evaporator while the supply pressure is controlled by the pressure control valves. The temperature of nitrogen gas supplied into the PCV is controlled at between 10°C and 50°C. The air in the PCV is exhausted by the PCV purge exhaust fans through the Reactor Area (R/A) HVAC.

The AC exhaust line isolation valves are closed when an inert condition in the PCV has been established.

(b) PCV Pressure Control Mode

The AC maintains the required slightly positive pressure inside the PCV to preclude air in-leakage from the secondary containment to the PCV by supplying nitrogen gas which compensates for the assumed leakage rate from the PCV (0.4%/d) during power operation.

During power operation, nitrogen gas is supplied from the AC Nitrogen Gas Supply Machine. The temperature of nitrogen gas generated from the AC Nitrogen Gas Supply Machine depends on the outside temperature and might be lower than 10°C which is the minimum operating temperature for normal operation of the PCV. Therefore, nitrogen gas is warmed up above 10°C by the AC Nitrogen Gas Heater.

Also, the AC maintains pressure in the PCV within the determined values by venting the gas as necessary for the following cases:

(i) Thermal expansion during plant start-up.
(ii) PCV pressure increase during normal operation due to leakage or discharge of gas from pneumatically controlled equipment.

The PCV exhaust lines are provided with small diameter bypass lines to facilitate PCV pressure control without sudden changes. PCV gases are normally exhausted through the R/A HVAC, but if increase of radioactivity in the exhaust gases is detected, the exhaust path is switched from the HVAC to the SGTS to treat the exhaust gases through the SGTS filter and reduce the radioactivity.

Exhaust gases are continuously monitored by the R/A HVAC and the SGTS flow meter and radiation monitor.

(c) De-Inerting Mode
The AC supplies the air and exhausts nitrogen gas from the PCV simultaneously when the plant is shut down to replace nitrogen with air and allow personnel access. The air to de-inert the atmosphere is supplied and exhausted by the R/A HVAC.

PCV gases are normally exhausted through the R/A HVAC, but if increase of radioactivity in the exhaust gases is detected, the exhaust path is switched from the HVAC to the SGTS to treat the exhaust gases through the SGTS filter and reduce the radioactivity.

(2) Equipment Design and Operation

(a) AC Nitrogen Gas Supply Facilities

(i) Purpose
The purpose of the AC Nitrogen Gas Supply Facilities is to supply nitrogen into the PCV to perform the inerting and the makeup operations ([AC SFC 4-17.1] [AC SFC 4-17.2] and [AC SFC 5-13.1]).

(ii) Configuration and Operation
The AC Nitrogen Gas Supply Facilities consist of the AC Liquid Nitrogen Evaporator, which supplies nitrogen gas for inerting operation, and the AC Nitrogen Gas Supply Machine, which supplies nitrogen gas for makeup during normal operation.

(iii) Performance
The AC Nitrogen Gas Supply Facilities are designed to supply a flow rate of inerting nitrogen of approximately 12,500m^3/h [normal] for the inerting operation, and a flow rate of approximately 48m^3/h [normal] for makeup during normal operation. The AC Nitrogen Gas Supply Facilities are designed to perform as shown in Table 13.3-3:

Table 13.3-3: AC Nitrogen Gas Supply Facilities Capacity

Inerting Capacity:	approx. 12500m^3/h [normal]
Makeup Capacity:	approx.48m^3/h [normal]

(b) AC Nitrogen Gas Heater

(i) Purpose
The AC Nitrogen Gas Heater is designed to warm the nitrogen gas supplied by the AC Nitrogen Gas Supply Machine as required ([AC SFC 4-17.1] [AC SFC 4-17.2] and [AC SFC 5-13.1]).

(ii) Configuration and Operation
The AC Nitrogen gas heater consists of the Electric Heater, which is put onto the pipe surface.

(iii) Performance
- Number: 1 unit
- Capacity: 1.6 kW

(3) Main Support Systems

(a) Instrumentation and Control

(i) Instrumentation
Instrumentation is provided to measure and monitor the operating conditions of the AC components necessary for the delivery of the safety functions. The main provisions for instrumentation are described as follows:

- PCV Oxygen Concentration
- PCV Pressure
- Nitrogen gas flow rate for inerting and Air flow rate for de-inerting

(ii) Control
The main control provisions related to the delivery of the safety functions are summarised as follows:

- The flow of nitrogen gas supply is automatically adjusted by the pressure control valves on the normal nitrogen supply line depending on pressure variations in the PCV in order to maintain the PCV pressure within the control targets.
- Opening of the isolation valves and operation of the exhaust fans are manually controlled from the MCR during de-inerting operation.

(b) Power Supply System
The configuration of the power supply system necessary to deliver the safety functions claimed is summarised as follows:
(i) The normal AC power supply to the AC system electrical components is provided by an independent off-site source (external grid).
(ii) In addition, the AC system is connected to emergency power sources to supply AC power and DC power to the valves, instruments and controllers necessary to deliver [AC SFC 3-2.1] and [AC SFC 4-7.1] in the event of LOOP.

(iii) The valves, instruments and controllers related to primary containment isolation [AC SFC 4-7.1] are provided with power supplied by Class 1 emergency power sources.
(iv) The valves, instruments and controllers related to containment venting [AC SFC 3-2.1] are provided with power by Class 2 emergency power sources.

(c) Heating Ventilating and Air Conditioning System

The HVAC provides a path for the drywell and wetwell exhaust flow during inerting, de-inerting, and venting, provides sufficient air flow to limit the concentration of any nitrogen leaking from the PCV into the secondary containment, and supplies air for purging the PCV during de-inerting.

(d) Standby Gas Treatment System (SGTS)

The SGTS processes any drywell bleed-off, inerting, and de-inerting exhaust flows, as required by offsite release constraints when radioactivity of the exhaust gas exceeds the specified values.

Assumptions, Limits and Conditions for Operations

In order to ensure that the PCV Gas Control Systems are operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCOs, 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 the BSC on PCV Gas Control Systems ([Ref-13.3-2]).

- Drywell pressure shall be within the prescribed limit during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.
- Oxygen concentration in PCV shall be within prescribed limit during power operation for the delivery of the SFCs claimed when required.
- Four PARs in the Drywell and one PAR in the Wetwell shall be operable during power operation and start-up for the delivery of the SFCs claimed when required.

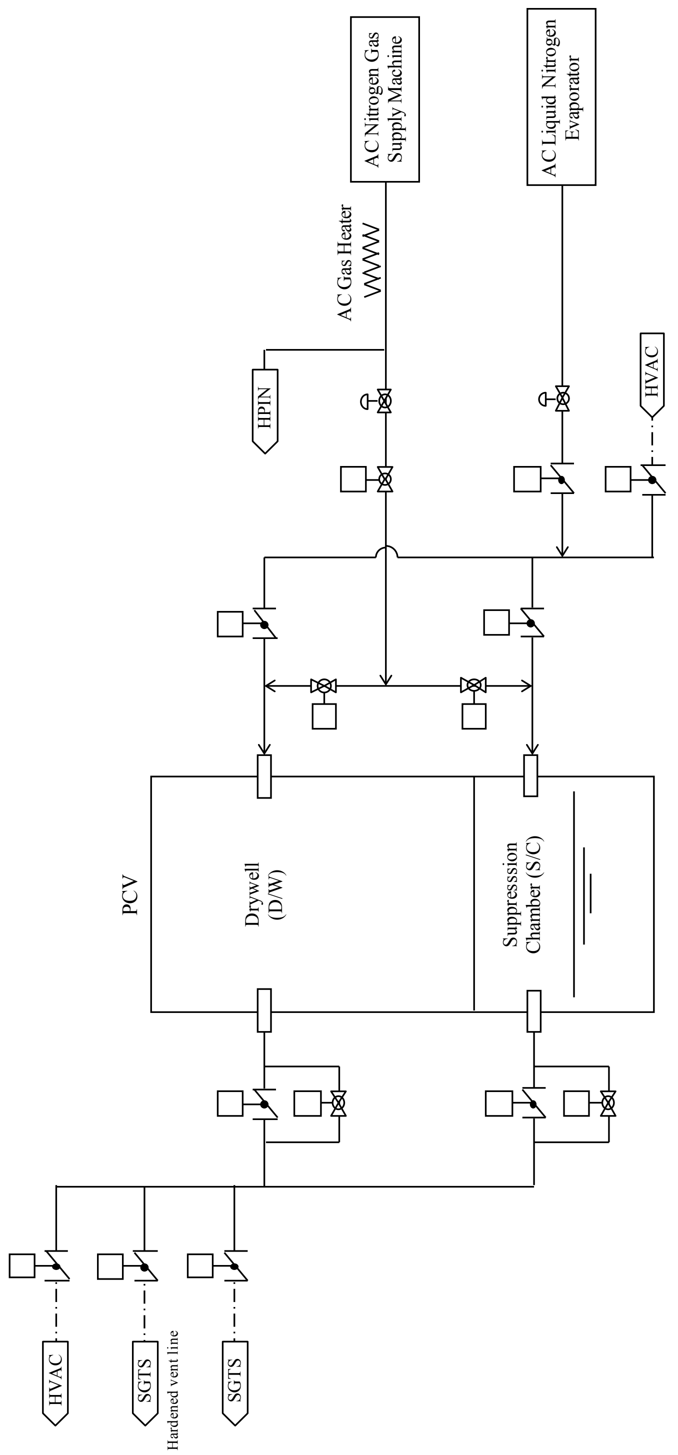

Figure 13.3-8: Outline of the Atmospheric Control System

13.3.3.4 Containment Heat Removal Systems

System Summary Description

This section is a general introduction to the Containment Heat Removal Systems where the systems roles, system functions and basic configuration are briefly described. The Containment Heat Removal Systems Safety Case is justified in the BSC on Containment Heat Removal Systems [Ref-13.3-3]. The systems which form it are described in the system specifications ([Ref-13.3-5] [Ref-13.3-8] and [Ref-13.3-12]) and the P&IDs ([Ref-13.3-6] [Ref-13.3-7] [Ref-13.3-9] [Ref-13.3-10] [Ref-13.3-11] and [Ref-13.3-13]).

(1) System Roles

The principal role of the Containment Heat Removal Systems is to remove from the Primary Containment Vessel (PCV) the decay heat generated by the core after reactor shutdown following a design basis fault in order to prevent excessive containment temperatures and pressure, thus maintaining containment integrity in the long term following a design basis fault.

The Containment Heat Removal Systems consist of the following Systems:

(a) The primary system for Containment Heat Removal is the RHR through its several operation modes: Low Pressure Flooder mode (LPFL), Suppression Pool Cooling mode (SPC) and PCV Spray cooling mode.

(b) The alternative means for Containment Heat Removal is the Containment Venting System (CVS) which consists of two subsystems that perform venting of the containment: the Atmospheric Control System (AC) and the Filtered Containment Venting System (FCVS). The detailed design of the AC is described in section 13.3.3.3 of this chapter about PCV Gas Control Systems and that of the FCVS is described in Chapter 16 : Auxiliary Systems, section 16.7 about Severe Accidents Mechanical Systems. This chapter only considers CVS SFCs for design basis faults whilst Chapter 16 also considers additional SFCs for the FCVS for beyond design basis faults and severe accidents.

(2) Functions Delivered

The Containment Heat Removal Systems are designed to perform the following functions:

(a) The RHR provides PCV cooling by encompassing several RHR operation modes: Low Pressure Flooding mode (LPFL), Suppression Pool Cooling mode (SPC) and PCV Spray Cooling mode.
(b) Furthermore, the RHR through the PCV sprays provides removal of fission products released in the containment (drywell) following a LOCA.
(c) The CVS releases non-condensable gases and steam to the main stack to prevent damage to the PCV due to overpressure in the event of failure of the primary heat removal means (RHR).

(3) Basic Configuration

The Containment Heat Removal Systems consist of the following main systems and components.

(a) RHR (S/P cooling mode, LPFL mode, PCV Spray mode)
Any one of the three divisions of the RHR is capable of delivering the S/P cooling and the LPFL, whilst two divisions can deliver the PCV Spray. For the three modes, the RHR pumps draw water from the S/P, circulate it through the RHR heat exchangers to remove heat and inject it to the S/P, the RPV, or the PCV Spray respectively for the three modes. The RHR components performing PCV heat removal are the RHR pumps (one per division), the RHR heat exchangers (two per division), the Suction Strainer on the suction lines inside the S/P, and the individual piping of each division (coolant suction line from the S/P and injection lines to each respective destination) as well as the necessary valves, instrumentation and controls.

(b) Containment Venting System (AC and FCVS)
The portion of the AC included in the CVS consists of the piping from the D/W and S/C outlets of the PCV to the hardened vent line, bypassing the SGTS to the main stack, as well as the necessary valves, instrumentation and controls. The FCVS consists of the piping from the D/W and S/C outlets of the PCV passing through the Vent Filter to the main stack as well as the necessary valves, instrumentation and controls.

(4) Design Bases

This section describes the design bases for the Containment Heat Removal Systems.

The Containment Heat Removal Systems have been designed to meet the following SFCs. The linkage between the SFCs of the Containment Heat Removal Systems with the FSFs and the HLSFs is shown in the Appendix A. The FSFs and HLSFs are defined in Chapter 5 : General Design Aspects.

<u>Fault Conditions</u>

[Primary means against design basis faults]

(a) 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.

(b) 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.

(c) 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.

(d) 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.

[Secondary means against design basis faults]

(e) The AC is a secondary means to deliver long term PCV heat removal and overpressure protection in the event of design basis faults where the primary long-term containment heat removal means (RHR) has failed. [AC SFC 3-2.1]

This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements.

(f) The FCVS is a secondary means to deliver long term PCV heat removal and overpressure protection in the event of design basis faults where the primary long-term containment heat removal means (RHR) has failed. [FCVS SFC 3-2.1]

This function is categorised as Safety Category A and the components to deliver it are designed to meet Safety Class 2 requirements.

System Design Description

This section describes the design of the Containment Heat Removal Systems to support and satisfy the delivery of [RHR SFC 3-1.3], [RHR SFC 3-1.4], [RHR SFC 3-1.5], [RHR SFC 4-7.2], [AC SFC 3-2.1] and [FCVS SFC 3-2.1]. Additional design description can be found in the system specifications ([Ref-13.3-5] [Ref-13.3-8] [Ref-13.3-12]) and the P&IDs ([Ref-13.3-6] [Ref-13.3-7] [Ref-13.3-9] [Ref-13.3-10] [Ref-13.3-11] [Ref-13.3-13]).

(1) System Design and Operation

(a) RHR modes of operation
The Containment Heat Removal System cooling capability encompasses several of the RHR operation modes, including the LPFL, the Suppression Pool Cooling and the PCV Spray Cooling (drywell and wetwell) modes.

Containment cooling starts as soon as the LPFL injection flow begins. The Suppression Pool Cooling mode removes containment heat by cooling S/P water. The containment sprays cool the drywell and wetwell by condensing steam and the condensate running back into the S/P. All water that leaves the S/P is cooled by the RHR Heat Exchangers during the three operational modes indicated above. For each of the three divisions that operate in LPFL mode, water is drawn from the S/P, pumped through the RHR Heat Exchangers and injected into the RPV. Also, for each of the three divisions that operate in S/P Cooling mode, water is drawn from the S/P, pumped through the RHR Heat Exchangers and released into the S/P. Two of the divisions of the RHR can operate in PCV Spray Cooling mode (RHR divisions B and C), in which water is drawn from the S/P, pumped through the RHR Heat Exchangers and sprayed through wetwell and drywell spray headers. A portion of the water returned to the S/P may be passed through wetwell spray headers. Figure 13.3-9, Figure 13.3-10 and Figure 13.3-11 show outlines of the RHR configurations for the three modes forming the Containment Heat Removal System.

Water from the Reactor Building Cooling Water System (RCW) is pumped through the heat exchanger shell side to exchange heat with the processed water.

The LPFL mode is automatically initiated from ECCS signals or manually initiated. The Suppression Pool Cooling mode is automatically initiated from S/P high temperature signal or manually initiated. The PCV spray cooling mode is manually initiated. The RHR must be realigned for Suppression Pool Cooling and PCV Spray cooling modes by the plant operator after the RPV water level has been recovered by the LPFL mode operation, but the RHR Pumps do not need to be actuated as they are already running. Suppression Pool Cooling mode is initiated in any of the three RHR divisions by closing the LPFL injection valve and opening the pool return valve. PCV Spray cooling mode is initiated in any of RHR divisions B or C by manually closing the LPFL injection valve and opening the spray valves.

The PCV Spray Cooling mode may be initiated manually only after a high drywell pressure permissive occurs (indicating a large LOCA inside PCV). The wetwell and drywell sprays are arranged such that sprayed water can be distributed equally and it does not impact directly on the walls and components surfaces. PCV spray cooling flow control valves are provided to adjust the flow manually by the operator.

In the event of the postulated LOCA, the short-term energy release from the reactor primary system is dumped to the S/P. Subsequent to the accident, fission product decay heat results in a continuing energy input to the pool. The Containment Heat Removal System removes this energy which is released into the primary containment system, thus maintaining acceptable S/P temperatures and containment pressures.

The containment spray features of the Containment Heat Removal System can reduce the amount of radioactive material released to the environment in the event core damage occurs. The additional benefits provided by the sprays are condensing steam and scrubbing of fission products in the containment atmosphere.

(b) PCV Venting operation

In the event of design basis faults combined with failure of the RHR to perform its functions (such as Station Blackout – SBO, which assumes LOOP and combined failure of the Safety Class 1 Emergency Diesel Generators), the long term heat removal function is performed by the CVS through venting of the gases contained inside the PCV. The gases are exhausted through the exhaust portion of the AC or through the FCVS.

The AC opens a pathway from the S/C airspace through the hardened vent line bypassing the SGTS filter trains to the main stack. The air-operated PCV isolation valves at the S/C and D/W outlets of the system are designed to be opened by dedicated air cylinder racks in case the Instrument Air System is lost. The FCVS releases PCV gas to the main stack for containment protection against overpressure and heat removal by opening the motor-operated valves at the PCV outlet. The released gas flow through the Vent Filter; however the filtering function is not needed in the case of design basis faults as it is assumed that no large release of radionuclides will happen. This assumption is supported by the analysis of design basis faults in Chapter 24 : Design Basis Analysis.

Venting operation also prevents further increase of the containment pressure and ensures it is maintained below the design values. Although exhaust lines are provided from the S/C and the D/W for both systems, for design basis faults, only the S/C exhaust lines valves are opened to make the gases pass through the Suppression Pool before they are released outside of the containment. This allows reducing radioactive content through scrubbing by S/P water.

Venting is performed until RHR functions are recovered (e.g. offsite power recovery) after which RHR is started up for long-term heat removal.

Figure 13.3-12 shows an outline of the Containment Venting System.

(2) Equipment Design and Operation

(a) RHR Heat Exchanger

The RHR Heat Exchanger is described in Chapter 12 : Reactor Coolant Systems, Reactivity Control Systems and Associated Systems section 12.3.5.4 related to the RHR.

(3) Main Support Systems

(a) Instrumentation and Control
The system supporting the RHR with instrumentation and control is the Safety Class 1 SSLC. The system supporting the venting function of the AC and the FCVS with instrumentation and control is the Safety Class 2 Hardwired Backup System (HWBS). The design and the claims on the SSLC and the HWBS are addressed in Chapter 14 : Control and Instrumentation.

(i) Instrumentation
Instrumentation is provided to measure and monitor the operating conditions of the system components necessary for the delivery of containment heat removal. The main provisions for instrumentation are described as follows:

Containment Heat Removal by the RHR:
- Temperature indicators are provided on the inlet and outlet of the RHR Heat Exchanger in order to monitor the heat exchanger performance and verify S/P temperature.
- The indication of temperature and water level inside the S/P are displayed and alarmed in the MCR.

Containment Heat Removal by the CVS:
- PCV Pressure
- D/W Temperature
- S/C Temperature

(ii) Control:

- The RHR is automatically initiated in LPFL mode upon LOCA signal (low reactor water level or high D/W pressure)
- The RHR Suppression Pool Cooling mode is automatically initiated upon detection of high S/P temperature.
- Transition to PCV Spray Cooling Mode:
  The D/W Spray line valves are capable of being opened only when the low pressure injection valves into the RPV are fully closed and when the drywell is in a high pressure condition, indicating a large LOCA inside the containment.
- PCV Venting:
  The CVS is initiated for venting operation by the operator by opening the PCV venting valves manually from the MCR. The valves are capable of being opened even if the containment isolation signal was transmitted.

(b) Power Supply System

Power supply for the RHR, components, valves, instrumentation and controllers come from the Safety Class 1 Electrical Power Distribution System (EPS). Power supply for the AC and the FCVS venting valves, and the related instrumentation and controllers are provided by the B/B Class 2 EPS. The detailed design and the claims on the EPS are addressed in Chapter 15 : Electrical Power Supplies.

(i) The RHR is connected to separated and independent divisions of the Safety Class 1 AC and DC EPS 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). The normal power supply to the RHR electrical components, valves, instrumentation and controllers is provided by the external grid. In addition, the Safety Class 1 Emergency Diesel Generators of Division I, II and III provide power for all RHR components in RHR division A, B and C in the event of LOOP

(ii) The CVS is connected to separated systems of the B/B Class 2 AC EPS supplying the required power to all electrical components in each subsystem (AC is connected to System 1 and FCVS is connected to System 2). The normal power supply to the CVS valves, instrumentation and controllers is provided by the external grid. In addition, the Class 2 B/B Generators A and B provide power to the CVS components in the event of LOOP or Station Blackout (SBO).

(c) Reactor Building Cooling Water System (RCW)

Cooling water for RHR components is supplied by the safety class 1 RCW. The detailed design and the claims on the RCW are addressed in Chapter 16 : Auxiliary Systems. Cooling water supply is not required for the CVS components.

(i) The RCW supplies water to the RHR Heat Exchangers, RHR Pumps, motors, bearings and seal water cooling equipment.
(ii) The RHR is connected to 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.

(4) System Architecture

(a) Redundancy

The RHR, which is the primary means for delivering containment heat removal, consists of three redundant divisions A, B, and C with their respective pumps, heat exchangers, strainers, piping and valves, and instrumentation and controls such that, single failure of any dynamic mechanical component does not prevent the delivery of the safety function. Likewise, RHR divisions B and C delivering PCV Spray Cooling mode provide redundancy for backup of containment heat removal, and each one is provided with their respective spray headers and valves.

The CVS, which delivers containment overpressure protection and heat removal as a secondary means, is provided with redundancy with two redundant subsystems (the AC and the FCVS) with their respective venting piping and valves, and instrumentation and controls such that single failure of any dynamic mechanical component does not prevent the delivery of the safety function.

(b) Independence

The three divisions of the RHR are functionally independent and physically separated in different locations to prevent failure of a component in one of the divisions from leading to a common cause failure of all divisions. Furthermore, the redundant supporting systems for each division of the RHR (C&I, power supply, cooling water and HVAC) are independent and separated as well.

The piping and valves of the CVS subsystems are functionally independent to prevent failure of a component in one of the systems from leading to a common cause failure of both systems; and physically separated from lower safety class items of which the failure could result in failure of the venting function. Furthermore, the redundant supporting systems for each division the venting systems (C&I, power supply, and HVAC) are independent and separated as well.

(c) Diversity

The design of the RHR and the CVS has been carried out taking into account diversity in terms of structure and components, operating conditions and functioning principles so that the risk of an event leading to a common failure of both the primary and the secondary means for containment heat removal is reduced. They are also physically separated to prevent an internal hazard from leading to their common failure. The same criteria are applied to the support systems (C&I, power supply and HVAC) so that failure of one of them does not result in full unavailability of the containment heat removal function.

Assumptions, Limits and Conditions for Operation

In order to ensure that the Containment Heat Removal Systems are operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCOs, 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 which is shown below is described in the BSC on Containment Heat Removal Systems ([Ref-13.3-3]).

- Suppression Pool temperature shall be within prescribed limits during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.
- Three divisions of RHR shall be operable for containment heat removal during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.
- Two divisions of CVS shall be operable during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.

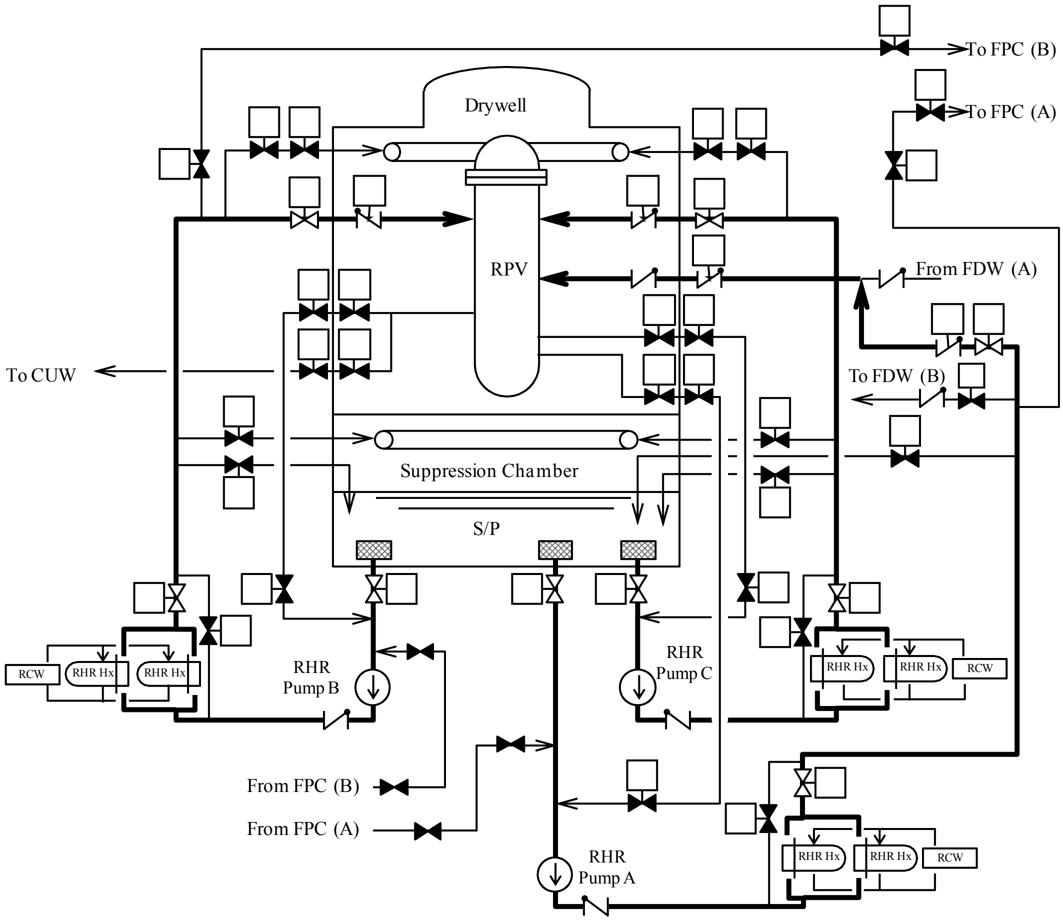

Figure 13.3-9: Outline of the LPFL Mode of RHR Operation

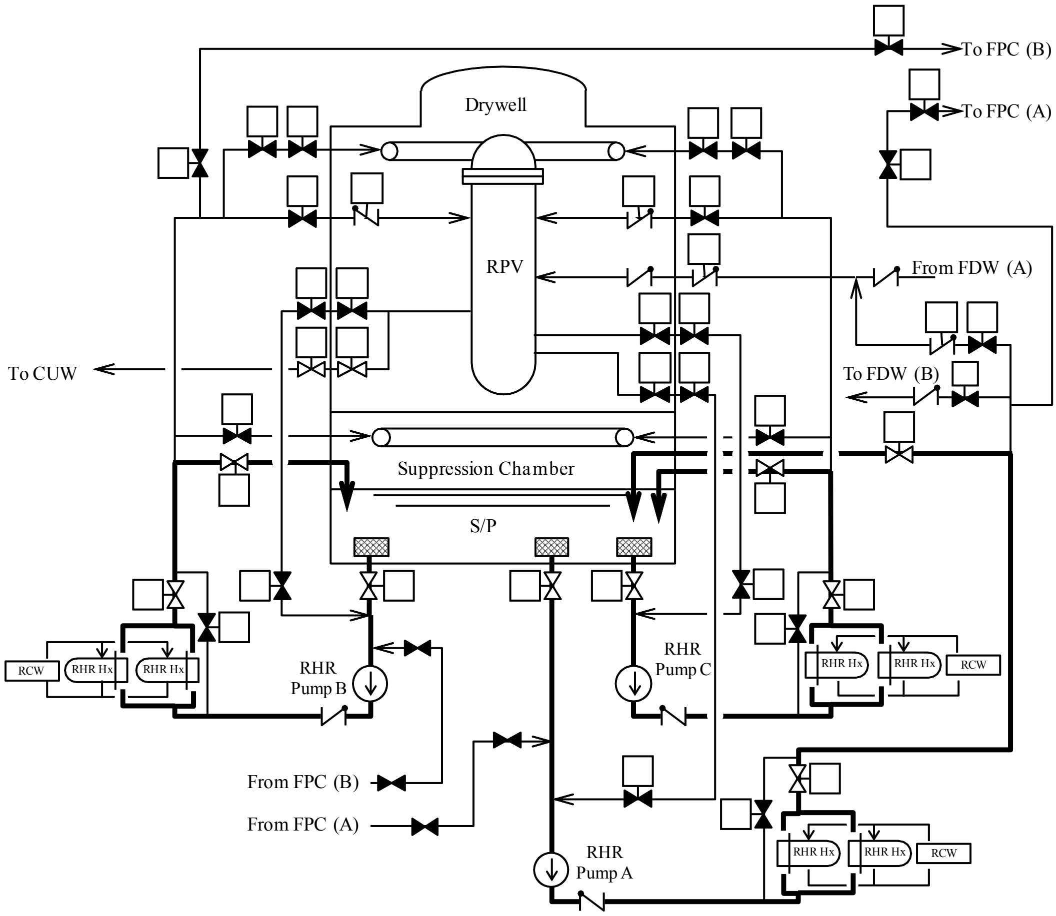

Figure 13.3-10 : Outline of Suppression Pool Cooling Mode of RHR Operation

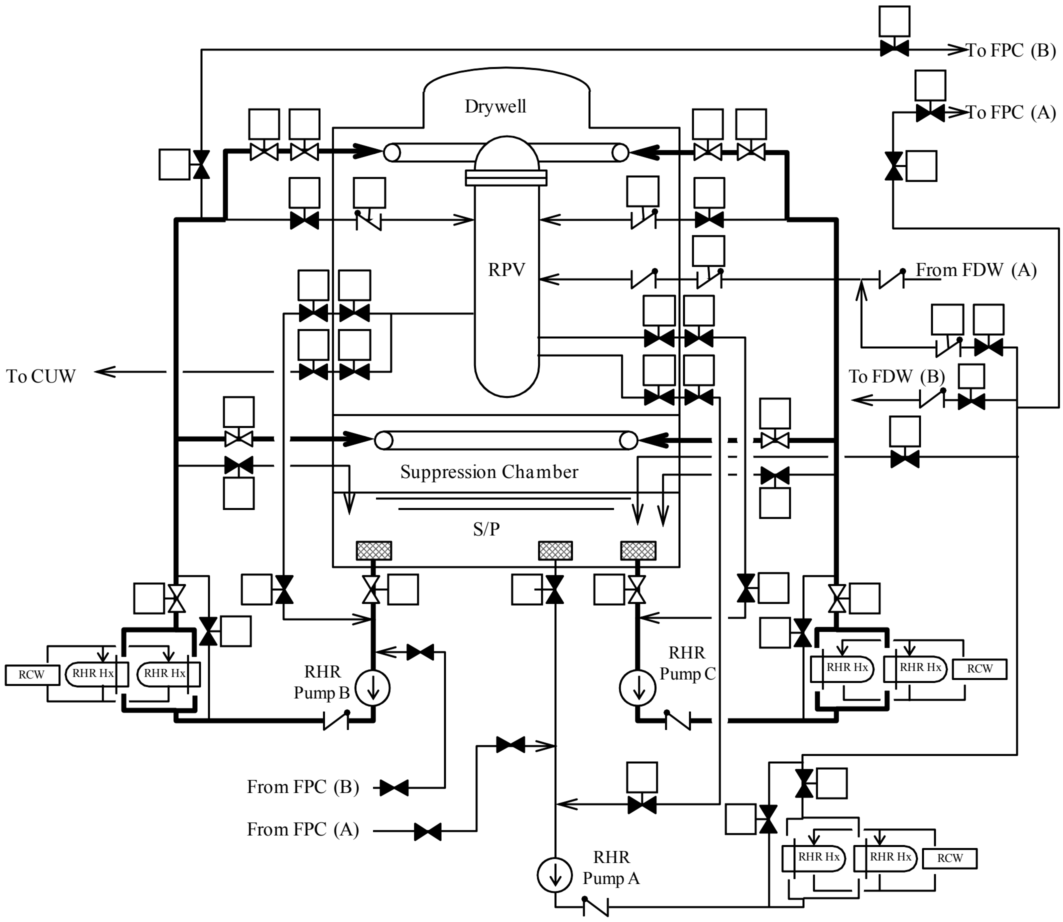

Figure 13.3-11: Outline of the PCV Spray Cooling Mode of RHR Operation

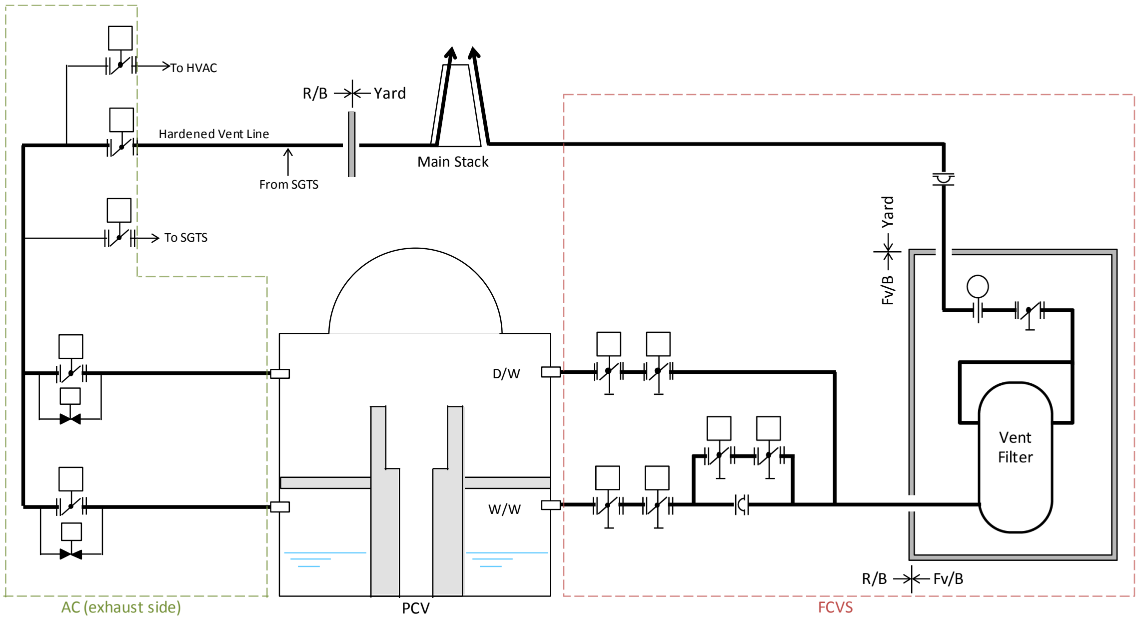

Figure 13.3-12: Outline of the Containment Venting System

13.3.3.5 Drywell Cooling System

This section is a general introduction to the Drywell Cooling System (DWC) where the system roles, system functions, system configuration and modes of operation are briefly described.

System Summary Description

(1) System Roles

The DWC is designed to achieve the following purposes:

(a) To maintain the required thermal environment and humidity so that the components in the drywell operate in a proper manner during normal plant operation and in the event of Loss of Off-site Power (LOOP).
(b) To cool the atmosphere in the drywell so that the working environment temperature during plant inspection and maintenance is acceptable for personnel access.

(2) Function Delivered

The DWC is designed to achieve the following functions:

(a) DWC Coolers and DWC Cooling Dehumidifiers supply the bulk head, inner drywell and pressure vessel skirt areas with cooled air (nitrogen during normal plant operation) through ducts.
(b) The circulating air / nitrogen is returned to the DWC Coolers once collected at the upper drywell area. Each DWC Cooler mixes, circulates and cools the air /nitrogen for each part of the drywell.
(c) The structures and components within the drywell are designed to withstand the Loss of Coolant Accident (LOCA) environmental conditions. Thus, in the event of a LOCA, the DWC function is not required.
(d) The cooling water for the DWC is isolated by motor operated valves in the event of a LOCA. Therefore, the DWC is shut down.

(3) Basic Configuration

The DWC consists of three DWC Coolers (50% × 3 units) supplying cooling air / nitrogen into each part of the drywell, two DWC Cooling Dehumidifiers (50% × 2 units), ducts, dampers, piping, valves, instrumentation and control devices, etc.

The DWC Cooler (one of which functions as a stand-by unit) consists of the DWC Supply Fan and Drywell Cooling Unit on which the cooling coils are mounted whereas the DWC Cooling Dehumidifier consists of the cooling coil only.

Figure 13.3-13 shows the basic DWC system configuration.

The SSCs of the DWC are categorised as Safety Category C and classified as Safety Class 3.

(4) Modes of Operation

(a) Normal operation mode
During normal operation, two units of both DWC Coolers and DWC Cooling Dehumidifiers are operated at the same time.
The RCW functions as a cooling medium for the cooling coil of the DWC Cooler which cools down the circulating air / nitrogen.

In addition, the cooling coil of the DWC Cooling Dehumidifier in the supply duct utilises the HVAC Normal Cooling Water System (HNCW) as cooling medium which cools and dehumidifies the supply air / nitrogen.

(b) Plant inspection and maintenance operation mode
Two units of DWC Cooling Dehumidifiers are operated to cool down the atmosphere (air) prior to and during plant inspection and maintenance activities. The cooling water for the cooling coil in the DWC Coolers can only be supplied at a higher temperature (constantly at 35 °C) than the required temperature within the PCV for the inspection and maintenance; Therefore only the supply fans of the DWC coolers are in operation in this mode, without the cooling water flow, with cooling supplied instead from the DWC Cooling Dehumidifiers.

(c) Loss of Off-site Power (LOOP) operation mode
In the event of LOOP not accompanied by Loss of Coolant Accident (LOCA), supply fans are automatically restarted after switching the power supply to the Emergency Diesel Generator System (EDG).

Design Bases

This section describes the design bases for the DWC. The linkage between the SFCs of the Drywell Cooling system with the FSFs and the HLSFs is shown in the Appendix A. The FSFs and HLSFs are defined in Chapter 5 : General Design Aspects.

(1) Safety Functions

The DWC has been designed to meet the following SFCs. The relationship between the SFCs put on this system and the high level claims is shown in Appendix-A.

<u>Normal Conditions</u>
(a) The DWC controls the design environmental parameters inside the served areas. [DWC SFC 5-18.1]

This function is classified 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 DWC to support and justify the delivery of [DWC SFC 5-18.1]. Additional design description can be found in [Ref-13.3-14] and [Ref-13.3-15].

(1) System Design and Operation

A stand-by DWC cooling unit is installed to enhance the reliability of the DWC system. The stand-by unit is automatically started when one of the duty units fails. Two dehumidifier coils with the capacity of 50% each are installed.

During normal operation, two units of both DWC Coolers and DWC Cooling Dehumidifiers are operated at the same time.

During plant inspection and maintenance mode, the cooling water for the cooling coil in the DWC Coolers can only be supplied at a higher temperature (constantly at 35 °C) than the required temperature within the PCV for the inspection and maintenance; therefore only the supply fans of the DWC coolers are in operation without the cooling water flow, with D/W cooling supplied instead from the DWC Cooling Dehumidifiers.

Fresh air for ventilation in the PCV during inspection and maintenance mode is introduced by the AC , as described in section 13.3.3.3.

In the event of LOOP not accompanied by LOCA, the DWC supply fans are automatically restarted after switching the power supply to the EDG.

(2) Equipment Design and Operation

(a) DWC Coolers (DWC Cooling Units and DWC Supply Fans)

(i) Purpose
The purpose of the DWC Cooler is to maintain the required thermal environment and humidity.

(ii) Configuration and Operation
The DWC Coolers (50% × 3 units, one of which functions as a stand-by unit) consist of DWC Cooling Units and DWC Supply Fans. The DWC Coolers provide cooled air / nitrogen within the PCV.

(iii) Performance
DWC Cooling Units
・ Unit Rated Air Flow approx. 60,000 m^3/h [per unit]
・ Cooling Coil Rated Air Flow approx. 60,000 m^3/h [per unit]
DWC Supply Fans
・ Rated Air Flow approx. 60,000 m^3/h [per unit]

(b) DWC Cooling Dehumidifiers

(i) Purpose
The purpose of the DWC Cooling Dehumidifier is to maintain the required thermal environment and humidity.

(ii) Configuration and Operation
The DWC Cooling Dehumidifiers (50% × 2 units) consist of cooling coils. DWC Cooling Dehumidifiers provide cooled air / nitrogen and remove humidity within PCV.

(iii) Performance
- Unit Rated Air Flow approx. 19,300 m^3/h [per unit]
- Cooling Coil Rated Air Flow approx. 19,300 m^3/h [per unit]

(3) Main Support Systems

The major support system related to delivery of the DWC safety functions claimed are briefly described as follows:

(a) Instrumentation and Control Systems
The principal objective for the method of control and instrumentation is to ensure the performance and reliability of the DWC system.

(i) Instrumentation
Instrumentation is provided to measure and monitor the operating conditions of the DWC components necessary for the delivery of the safety functions. The main provisions for instrumentation are described as follows.

- DWC cooling unit inlet (measured and monitored by AC)
- DWC cooling unit outlet (measured and monitored by AC)
- DWC dehumidifier coils outlet (measured and monitored by AC)

(ii) Control
The main control provisions related to the delivery of the safety functions are described as follows.
- The operation and monitoring of the system is carried out by use of flat displays installed in the MCR.
- A stand-by DWC Supply Fan starts up automatically when the Supply Fan trips.
- DWC Supply Fans are supposed to stop in the event of LOOP not accompanied by LOCA, however they are to be restarted automatically after activating power supply by EDG.
- DWC Supply Fans stop automatically when receiving the LOCA signal.

(b) Power Supply Systems

The normal AC power supply to the DWC electrical components is provided by an independent and off-site source (external grid). In addition, the DWC Supply Fans are provided with emergency AC power supply.

In the event of LOOP the DWC Supply Fans are supplied power by the EDG.

(c) Reactor Building Cooling Water System (RCW)
The RCW provides cooling water to the DWC Coolers.

(d) HVAC Normal Cooling Water System (HNCW)
The HNCW provides cooling water to the DWC Cooling Dehumidifiers.

Assumptions, Limits and Conditions for Operation

In order to ensure that the DWC is operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCOs, 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 which is shown below is reflected in the generic technical specifications [Ref-13.1-5].

- Drywell air temperature shall be within limits during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.

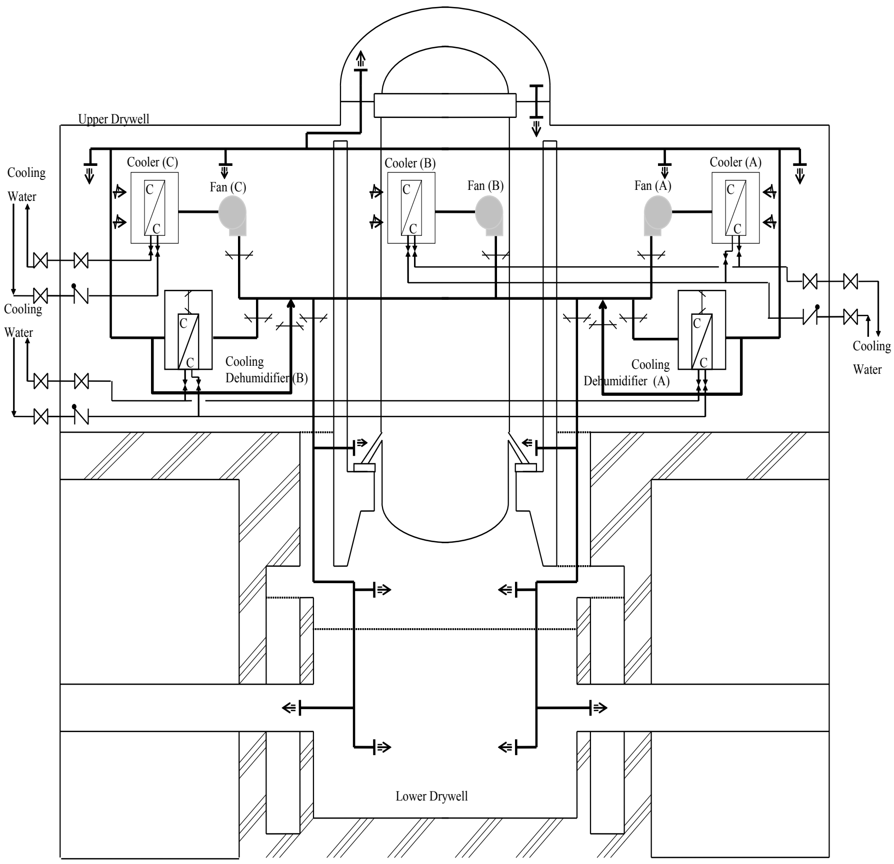

Figure 13.3-13: DWC System Configuration

13.3.4 Secondary Containment Facility

13.3.4.1 Reactor Building

(1) Facility Summary Description

The secondary containment boundary completely surrounds the PCV except for the basemat, which is common to both and, together with the clean zone, comprises the R/B as shown on Figure 13.3-14. The secondary containment encloses all penetrations through the PCV and all those systems external to the PCV that may become a potential source of radioactive release after an accident, including the main steam tunnel.

During normal plant operation, the secondary containment areas are kept at a negative pressure with respect to the environment and clean zone by the HVAC. Following an accident, the SGTS provides this function. These systems are described in Generic PCSR Chapter 16 : Section 16.5 related to the HVAC and section 13.3.4.2 of this chapter related to the SGTS.

Radioactive substances (fission products, activation products) that may, following an accident, leak from the primary to secondary containment are processed and captured by the SGTS, which ensures adequate delay time to allow radioactive decay before discharge to the environment. The HVAC exhaust systems and SGTS are located within the secondary containment to assure collection of any leakage. The secondary containment provides detection of the level of radioactivity released to the environment during abnormal and accident plant conditions. Personnel or material entrances to the secondary containment consist of airlocks with interlocked double doors or hatches and all the penetrations are sufficiently sealed providing a high degree of air tightness in the secondary containment.

(2) Facility Design Bases

The secondary containment is provided to confine and collect radioactive substances which may leak from the primary containment following a DBA. This collection allows effective filtration by the SGTS prior to release to the environment. The secondary containment region completely surrounds the PCV.

Adequate instrumentation and indications in the Main Control Room (MCR) are provided to monitor all important secondary containment parameters.

The secondary containment is periodically tested in conjunction with the SGTS to assure that the required vacuum can be maintained so that confinement performance is satisfied. [Ref-13.1-5]

(3) Facility Design

The secondary containment is a reinforced concrete building that forms an envelope surrounding the PCV above the basemat. The secondary containment has isolation systems on piping, doors and other penetrations. Details of structural design and arrangement of compartments for various systems are described in Chapter 10 : Civil Works and Structures.

During normal operation, the secondary containment system is operated at a slightly negative pressure relative to the atmosphere by the HVAC. This assures that any leakage from the PCV is collected and can be treated before release if its radioactivity level is above prescribed limits.

The building effluents are monitored for radioactivity. If the level of radioactivity rises above set levels, the secondary containment discharge can be routed through the SGTS, which incorporates High Efficiency Particulate Air (HEPA) filters and charcoal beds to remove radioactivity before release.

During normal operation, the secondary containment is the envelope that forces collection of airborne radioactive material from the Fuel Storage Pools, CUW, Fuel Pool Cooling and Clean-up system (FPC), Suppression Pool Clean-up system (SPCU) and other potentially radioactive sources in the secondary containment. The HVAC exhaust systems and SGTS are also located within the secondary containment to assure collection of any leakage. The RHR and HPCF pump seals and valve packings and RCIC components are a potential source of radioactive release and are located within the secondary containment.

During refuelling operations, once the PCV nitrogen atmosphere has been removed, the drywell head is removed and the secondary containment becomes the containment envelope. Therefore, entry into the secondary containment is provided via double door vestibules or, in the case of the main equipment hatch, a double door entry. This assures the integrity of the secondary containment envelope with effluent monitoring and treatment of airborne radioactive material resulting from normal plant or refuelling operations or from abnormal events such as a fuel drop accident.

Automatic shutoff of the normal HVAC air supply and ventilation exhaust of other systems after a LOCA or on detection of high radiation in the effluent is provided to prevent airborne leakage from escaping the secondary containment.

The airborne fission products are contained by maintaining all portions of the secondary containment at a negative pressure of approximately 60Pa [dif] relative to the lowest pressure boundary outside the secondary containment. This negative pressure is achieved following an accident by the SGTS. The airborne fission product leakage from the PCV is processed by the SGTS. The SGTS removes radioactive substances from the effluent gases with an efficiency of approximately 99.99% for iodine and approximately 99.9% for other radioactive substances prior to discharge to the environment.

All effluents processed by the SGTS from the secondary containment areas are monitored for gamma radiation level prior to their release to the environment.

All openings through the secondary containment boundary, such as personnel and equipment doors, remain closed after a DBA by interlocks or administrative control. These doors are provided with position indicators and alarms having readout and annunciation capability in the MCR.

The liquid leakage from the secondary containment to the clean zone or the environment is controlled, as required, by means of water loop seals, automatic shutoff valves in series, or piping upgrade to safety class. All system operations that transport liquid from the secondary containment to the clean zone or to the environment is automatically shut off during an accident and can only be manually initiated following an accident.

A high-energy pipe break within the secondary containment is detected by the Leakage Detection System (LDS) that is described in Chapter 12 : Reactor Coolant Systems and Associated Systems and Reactivity Control systems. Blowout panels are provided where necessary to relieve the thermal and pressure build-up in the various sub-compartments.

Periodic leakage tests to verify the air tightness of the secondary containment are carried out. The tests are performed by isolating the normal HVAC and operating the SGTS to maintain a negative pressure within the secondary containment.

Assumptions, Limits and Conditions for Operations

In order to ensure that the Secondary Containment is operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCO, 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 reflected in the generic technical specifications [Ref-13.1.5].

- Secondary Containment shall be operable during power operation, start-up and hot shutdown, during movement of irradiated fuel assemblies in the secondary containment, and during core alterations for the delivery of the SFCs claimed when required.

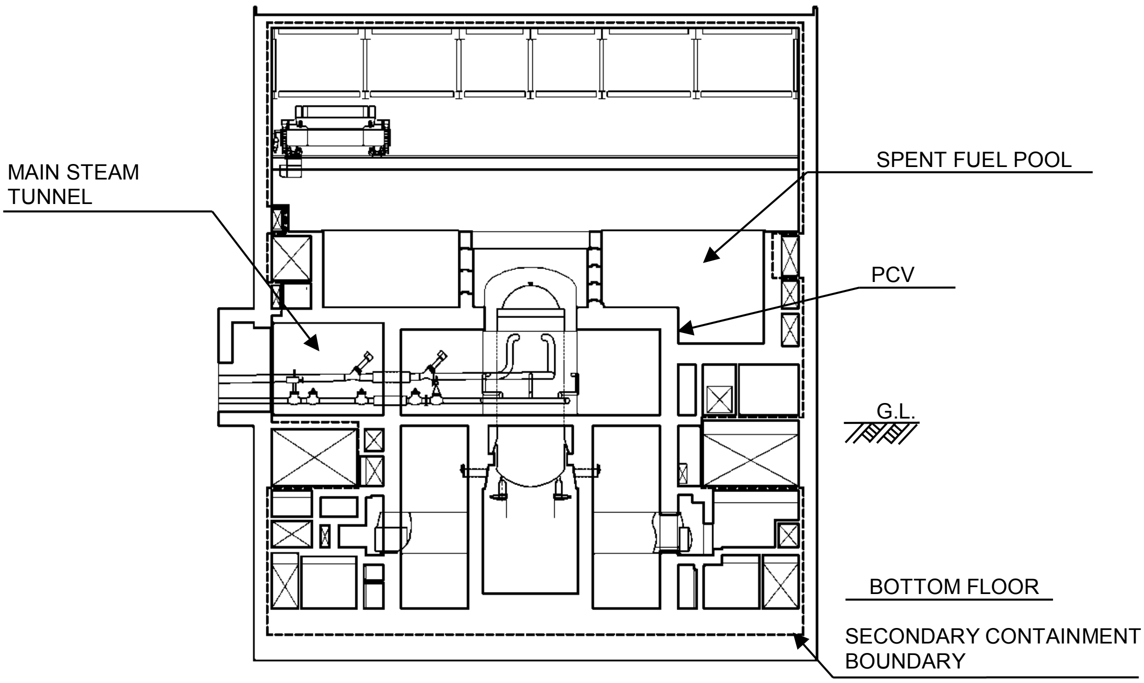

Figure 13.3-14: Conceptual Configuration of the Secondary Containment

13.3.4.2 Standby Gas Treatment System

System Summary Description

This section is a general introduction to the SGTS, in which the system roles, system functions, system configuration and modes of operation are briefly described. The SGTS safety case is justified in the BSC on SGTS [Ref-13.3-4]. The SGTS is described in detail in the system specifications [Ref-13.3-16] and the P&ID [Ref-13.3-17].

(1) System Roles

The main roles of the SGTS are to maintain the Secondary Containment areas under a negative pressure with respect to the environment following an accident and to treat radioactive substances that may leak from the Primary Containment to the Secondary Containment before being discharged to the environment.

(2) Functions Delivered

The SGTS is designed to perform the following safety functions:

(a) The SGTS controls the emission of radioactive substances by maintaining a negative pressure in the Secondary Containment and by filtering the effluent prior to discharge to the atmosphere following a Loss of Coolant Accident (LOCA) or a fuel handling accident.
(b) The SGTS processes gaseous effluent from the Primary Containment Vessel (PCV) and the Secondary Containment when it is required to limit the discharge of radioactivity to the environment during normal and abnormal plant operation.

(3) Basic Configuration

The SGTS consists of two divisions including a fan and a filter train each as well as the necessary piping, valves, instruments and controllers. The main components are summarised as follows:

(a)	SGTS Fan:	2 units
(b)	SGTS Filter Train:	2 units
(c)	SGTS Filter Train consists of the following components (per one SGTS Filter Train unit):	
	・ Moisture separator	1 set
	・ Process Heater	1 set
	・ Pre-filter	1 set
	・ HEPA filters	2 sets
	・ Charcoal filter	1 set
	・ Space Heaters	1 set
	・ Piping and valves:	1 set
	・ Instruments and controllers:	1 set
	・ Control panel:	1 set

Figure 13.3-15 shows an outline of the SGTS basic configuration.

(4) Modes of Operation

The SGTS can deliver the following three operation modes:

(a) Standby Mode
The SGTS is on standby during normal plant operation. The space heaters are operated during standby mode to heat the air around the charcoal filters and reduce the relative humidity in order to ensure the charcoal filter efficiency.

(b) Emergency Operation Mode
Upon the receipt of a safety initiation signal, the SGTS automatically starts operating to maintain a negative pressure in the Secondary Containment in order to control the discharge of radioactive substances, and process the gaseous effluent through the filter train to remove any airborne iodine and particulates.

(c) PCV Gas Exhaust Operation Mode
The SGTS is capable of manual initiation to discharge the gases inside the PCV if necessary, for example, during de-inerting of the PCV prior to refuelling outages depending on the radiation levels detected in the PCV gases.

(5) Design Bases

This section describes the design bases for the SGTS.

The SGTS has been designed to meet the following SFCs. The linkage between the SFCs of the SGTS with the FSFs and the HLSFs is shown in the Appendix A. The FSFs and HLSFs are defined in Chapter 5 : General Design Aspects.

<u>Normal Conditions</u>

(a) The SGTS constitutes a part of the confinement function by filtering the gaseous effluent from the Primary Containment or from the Secondary Containment when required to limit the discharge of radioactive material to the environment. [SGTS SFC 4-7.1]

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 SGTS constitutes a part of the confinement function in the event of design basis faults such as LOCA, radioactive releases from refuelling operations, etc. by maintaining a negative pressure in the Secondary Containment relative to the outdoor atmosphere, and by filtering radiological effluents from the Primary Containment that leak into the Secondary Containment to control and reduce the release of radioactive substances to the environment. [SGTS 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.

System Design Description

This section describes the design of the SGTS to support and justify the delivery of SGTS SFC 4-7.1 and SGTS SFC 4-7.2. Additional design description can be found in [Ref-13.3-4], [Ref-13.3-16] and [Ref-13.3-17].

(1) System Design and Operation

The SGTS is composed of two redundant fan and filter trains located in two adjacent rooms. Suction is taken from the Secondary Containment, including above the refuelling area, or from the primary containment via the AC. The treated discharge goes to the main plant stack. The main SGTS equipment is located within the Secondary Containment boundary.

Upon receipt of a high drywell pressure signal or a low reactor water level signal, or when high radioactivity is detected in the Secondary Containment or refuelling floor ventilation exhaust (initiation signals are indicated in section (3) “Main Support Systems”), both SGTS trains are automatically actuated and one train is manually placed in standby when the operation of both the trains is confirmed by the operator.

During normal operation, the SGTS filters the gaseous effluent from the Primary Containment or from the Secondary Containment when required to limit the discharge of radioactive material to the environment by manual actuation from the MCR. In the event of design basis faults, the SGTS maintains sufficient negative pressure (approximately 60Pa [dif]) in the Secondary Containment and filters radiological effluents from the Primary Containment that leak into the Secondary Containment to control and reduce the release of radioactive substances to the environment by automatic initiation.

(2) Equipment Design and Operation

(a) SGTS Filter Train

(i) Purpose
The SGTS Filter Train filters the gaseous effluent from the Primary Containment and/or the Secondary Containment for the delivery of [SGTS SFC 4-7.1] and [SGTS SFC 4-7.2].

(ii) Configuration and Operation
Each SGTS division is provided with one 100%-capacity SGTS Filter Train of approximately 2,000m^3/h of rated capacity. Each SGTS Filter Train consists of a moisture separator, process heater, filters, space heaters, filter housings, drain piping, upstream and downstream valves, and instrumentation. The four filters installed within the SGTS Filter Train are arranged with the following order from the upstream side: a Pre-Filter, a primary HEPA filter, Charcoal Filter and the secondary HEPA filter. Casings are welded structures to ensure leak tightness. The main configuration aspects with regard to the SGTS Filter Train are described as follows:

- Moisture Separator
  The Moisture Separator removes water droplets from the exhaust gases.
- Process Heater
  The process heater is provided to prevent degradation of the Charcoal and HEPA filters due to moisture.
- Pre Filter

The Pre-filter is provided to prevent a decrease in efficiency of HEPA filter due to clogging.

- HEPA Filter
  The HEPA filter located on the upstream side of the Charcoal filter prevents degradation of the Charcoal filter performance due to radioactive particulates that have leaked into the Secondary Containment, as well as preventing the release of these particles outside the Secondary Containment.
- The HEPA filter located on the downstream side of the Charcoal filter captures charcoal particles that adsorb radioactive iodine.
- Charcoal filter
  The charcoal filter removes radioactive iodine with an efficiency above 99.99% for inorganic and organic iodine.
- Space heater
  The space heaters are provided to prevent the degradation of the charcoal capability to adsorb iodine due to moisture in the air while the SGTS is on standby.

(iii) Performance
The SGTS Filter Train is designed to perform as follows in order to delivery [SGTS SFC 4-7.1] and [SGTS SFC 4-7.2]:

Table 13.3-4: SGTS Filter Train Parameters

SGTS Filter Train Specifications	
Number:	2 units
Capacity:	approx. 2000m^3/h per unit
HEPA Filter	
Number:	2 sets (2 filters/set: 4 filters in total)
Removal efficiency:	> 99.9%
Charcoal Filter	
Number:	1 set
Removal efficiency:	> 99.99% iodine

(b) SGTS Fan

(i) Purpose
The SGTS Fan takes suction of the gaseous effluent from the Primary Containment and/ or the Secondary Containment in order to deliver [SGTS SFC 4-7.1] and [SGTS SFC 4-7.2].

(ii) Configuration and Operation
The SGTS is provided with two redundant 100%-capacity SGTS Fans with a rated air flow of approximately 2,000m^3/h each. The SGTS Fan is located upstream of the SGTS Filter Train to prevent any gases which may leak in via fan shaft seals from bypassing the filter trains

(iii) Performance
The SGTS Fan is designed to perform as follows in order to deliver [SGTS SFC 4-7.1] and [SGTS SFC 4-7.2]:

Table 13.3-5: SGTS Fan Parameters

SGTS Fan Specifications	
Number:	2 units
Type:	Centrifugal
Capacity:	approx. 2000m^3/h per unit

(3) Main Support Systems

(a) Instrumentation and Control

(i) Instrumentation
Instrumentation is provided to measure and monitor the operating conditions of the SGTS components necessary for the delivery of the safety functions. The following parameters are displayed and/or recorded and alarms are initiated in the MCR. Moreover, the operating status of SGTS Fans and the valve operating status are displayed in the MCR as well.

- SGTS exhaust flow rate
- SGTS Filter Train inlet temperature, SGTS Charcoal Filter inlet temperature, SGTS Charcoal Filter outlet temperature and SGTS Charcoal Filter temperature
- Differential pressure between R/B pressure and atmosphere

(ii) Control
The main control provisions related to the delivery of the safety functions are summarised as follows:

- Initiation Signals
  The SGTS is automatically initiated upon any of the following signals in the event of a reactor accident, and also designed to be initiated manually from the MCR:

  - High radioactivity of the HVAC discharge gas in the Secondary Containment
  - High radioactivity in fuel handling area
  - Drywell high pressure
  - Reactor water low level

  Moreover, SGTS is automatically initiated upon Reactor Area HVAC failure signal even if no accident conditions.

- System Shutdown Signal
  The SGTS is shut down from the MCR by manual shutdown signal.

(b) Power Supply System

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

(c) Atmospheric Control System (AC)

The SGTS is manually initiated to process primary containment gaseous flow from the AC when required.

(4) System Architecture

(a) Redundancy
The SGTS consists of two 100%-capacity redundant divisions with their respective fans, filter trains, piping, valves, and instrumentation such that, single failure of any dynamic mechanical component does not prevent the delivery of the safety functions.

(b) Independence
The components of the two trains of the SGTS are independent and physically separated in adjacent rooms within the Secondary Containment in the R/B to prevent failure of a component in one of the divisions from leading to a common cause failure of the other train.

Assumptions, Limits and Conditions for Operation

In order to ensure that the SGTS is operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCOs, 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 the BSC on Standby Gas Treatment System ([Ref-13.3-4]).

- Two trains of the SGTS trains shall be operable during power operation, start-up, hot shutdown and during movement of irradiated fuel assemblies in the secondary containment.

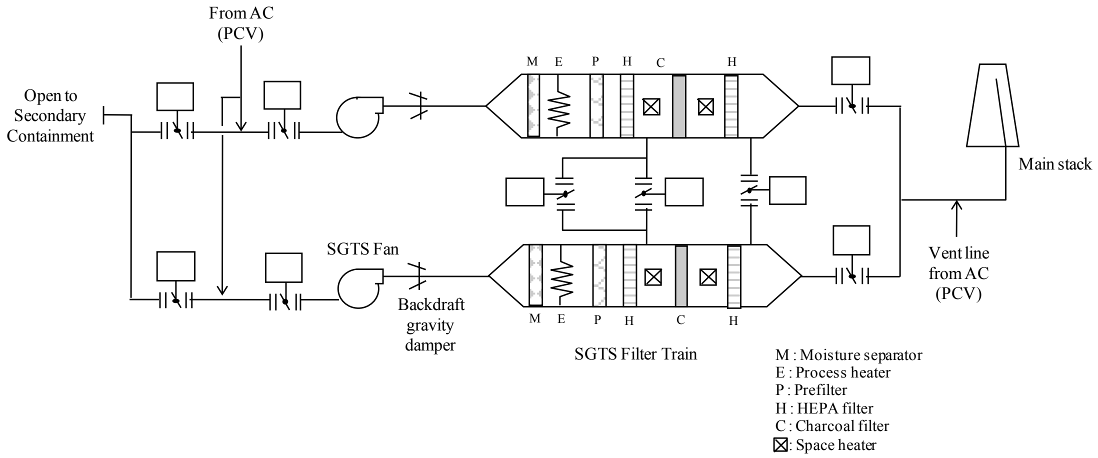

Figure 13.3-15: Outline of the Standby Gas Treatment System

13.3.5 Safety Design Evaluation

13.3.5.1 Analysis of Pressure and Temperature Responses of Primary Containment Vessel

In order to confirm the integrity of the primary containment vessel during a LOCA, an instantaneous double-ended break of the feedwater line, in which the pressure of the PCV is maximized, is analyzed as the bounding design basis accident for the primary containment vessel integrity. Immediately following a double-ended break in one of the two main feedwater lines just outside the reactor pressure vessel (Figure 13.3-16), the flow from both sides of the break will be limited to the maximum allowed by critical flow considerations.

Also, an analysis is conducted of a complete break accident of the main steam line, in which the temperature of the PCV is maximized. The main steam line break is a double-ended break with one end fed by the RPV directly through the broken line, and the other fed by the RPV through the unbroken main steam lines until the MSIVs are closed. (Figure 13.3-17)

When a LOCA occurs, the coolant flows out rapidly from the reactor and turbine side into the D/W and the D/W pressure increases. For this reason, most of the gases inside the D/W are driven out by the outgoing flow of reactor coolant into the S/C, and the steam in the gases is condensed by the S/P water. On the other hand, the non-condensable gases migrate to the airspace of the S/C, resulting in an increase of pressure in the S/C.

After the water level in the RPV has been restored to the elevation of the feedwater lines by the activation of the ECCS, the excess water will flow out through the break into the D/W. It cools and condenses the steam in the D/W and causes the heat generated in the core to move into the S/C. As a result of condensation of the steam in the D/W, the D/W pressure decreases, and the vacuum breakers activate to redistribute the non-condensable gases in the S/C to the D/W and the S/C. The RHR System is used at first as a Low-Pressure Flooder System to refill the RPV without operation of the RHR heat exchanger. However, 30 minutes after the initiation of the accident, heat removal by the RHR heat exchanger becomes available.

After heat removal from the S/C by the RHR System has begun and the amount of heat generated from the core has become equal to the amount of heat removed by the cooling system, the temperature in the D/W and the S/C starts to drop, and the pressure also falls.

The results of the analyses confirm that the maximum pressure and maximum temperature are kept below the specified design basis analysis acceptance criteria during LOCA events. Details of the analysis are shown in Chapter 24 : Design Basis Analysis, in subsection 24.8.3.

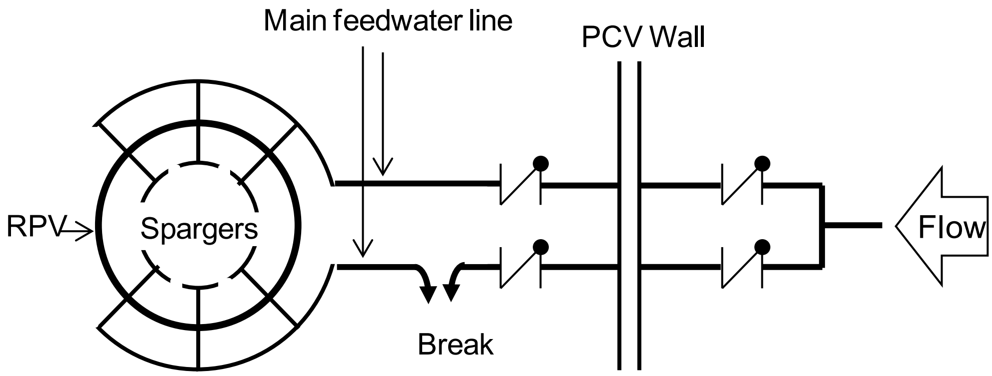

Figure 13.3-16: A Double-ended Break in a Feedwater Line

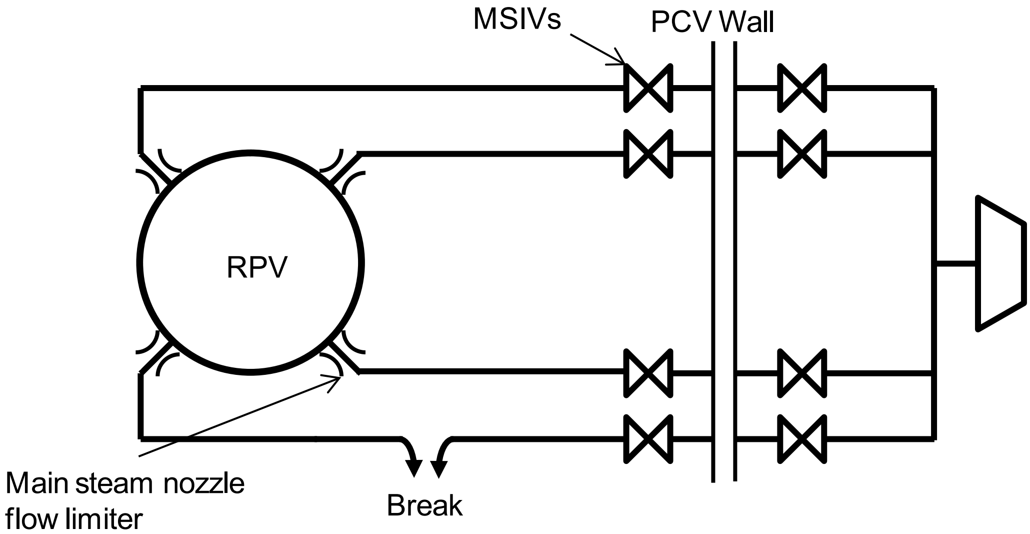

Figure 13.3-17: A Double-ended Break in a Main Steam Line

13.3.5.2 Analysis of Generation of Flammable Gases

Following a design basis LOCA, hydrogen and oxygen are generated by radiolytic decomposition of the water in the reactor core. The ABWR features Passive Auto-catalytic Recombiners (PARs) in the primary containment, which operate by passive means to reduce the concentrations of hydrogen and oxygen, in order to avoid the gradual accumulation of hydrogen and oxygen in an isolated containment. The PARs prevents the concentration of the gases exceeding a flammability limit, which if exceeded could lead to a deflagration and a resulting significant increase of temperature and pressure in the containment. For reference, various types of driving force in the containment can be expected to sufficiently mix the flammable gases. Transients in the concentrations of the flammable gases in the containment during a LOCA are analysed.

Following a design basis LOCA, generated hydrogen and oxygen are released into the drywell through the break. Therefore, concentration of hydrogen and oxygen in the D/W gradually increase. When the hydrogen concentration reaches 1.5 vol%, the PARs passively start to recombine hydrogen and oxygen, and if the hydrogen concentration subsequently falls to below 1.5 vol%, recombination stops until the hydrogen concentration exceeds 1.5 vol% again.

The results of the evaluation confirm that the concentration of flammable gases are kept lower than the combustion limit values of 4 vol% for hydrogen and 5 vol% for oxygen. Details of analysis are shown in Chapter 24 : Design Basis Analysis section 24.8.3.

13.4 Emergency Core Cooling System

13.4.1 System Summary Description

This section is a general introduction to the ECCS and the alternative systems for core cooling, in which the system roles, system functions, system configuration and modes of operation are briefly described. The ECCS and the alternative systems for core cooling safety case is justified in the BSC on Emergency Core Cooling System [Ref-13.4-1]. The ECCS design is described in detail in the system specifications ([Ref-13.4-2] [Ref-13.4-5] [Ref-13.4-8] [Ref-13.4-12] [Ref-13.4-14] and [Ref-13.4-17]) and the piping and instrumentation diagrams ([Ref-13.4-3] [Ref-13.4-4] [Ref-13.4-6] [Ref-13.4-7] [Ref-13.4-9] [Ref-13.4-10] [Ref-13.4-11] [Ref-13.4-13] [Ref-13.4-15] [Ref-13.4-16] and [Ref-13.4-18]).

(1) System Roles

The main role of the ECCS and the alternative systems for core cooling is to inject water inside the Reactor Pressure Vessel (RPV) in the event of a reactor fault such as loss of the main condenser, Loss of Coolant Accident (LOCA), Loss of Offsite Power (LOOP), etc., in order to prevent or suppress the zirconium water reaction and prevent serious damage to the core fuel.

(2) Functions Delivered

The ECCS is designed to perform the following functions:

(a) The Reactor Core Isolation Cooling System (RCIC) provides core cooling water supply to the RPV when the reactor is in a high pressure condition to compensate for water loss during transients and LOCA events.

(b) The High Pressure Core Flooder System (HPCF) provides core cooling water supply to the RPV when the reactor is in a high pressure or low pressure condition to compensate for water loss during transients and LOCA events.

(c) The Low Pressure Flooder (LPFL) Mode of the RHR provides core cooling water supply to the RPV when the reactor is in a low pressure condition to compensate for water loss and remove decay heat in the event of LOCA.

(d) The Automatic Depressurization System (ADS) depressurises the RPV to allow operation of the LPFL.

In addition, the alternative systems described below are provided in to perform similar functions in the event the ECCS failed to perform its functions.

(e) The Flooder System of Specific Safety Facility (FLSS) provides core cooling water supply to the RPV when the reactor is in low pressure condition in the event of failure of the primary cooling means (RCIC, HPCF and LPFL).

(f) The Reactor Depressurisation Control Facility (RDCF) depressurises the RPV to allow low-pressure water injection into the RPV with the FLSS.

The FLSS and RDCF detailed design is presented in Chapter 16 : Auxiliary Systems section 16.7 about Severe Accidents Mechanical Systems. This chapter only considers

FLSS and RDCF SFCs for design basis faults whilst Chapter 16 also considers additional SFCs for the FLSS and the RDCF for beyond design basis faults and severe accidents.

(3) Basic Configuration

The ECCS and the alternative systems for core cooling consist of the following subsystems and main components. Figure 13.4-1 shows an outline of the configuration of the ECCS and alternative systems for core cooling.

ECCS Configuration

The ECCS network consists of three independent divisions, I, II, and III. Each division has a high pressure and low pressure water injection function into the RPV. The ADS SRVs operate in conjunction with all divisions. The configuration is summarised as follows:

(i) Division I: RCIC + LPFL (A)
(ii) Division II: HPCF (B) + LPFL (B)
(iii) Division III: HPCF (C) + LPFL (C)
(iv) All divisions: SRVs (SRVs with safety/relief function and ADS)

(a) RCIC
The RCIC consists of a turbine-integrated pump which performs coolant injection into the reactor core in the event the RPV is isolated and the Feedwater (FDW) supply is unavailable. The turbine is driven by steam supplied from the RPV so that the pump can inject coolant into the RPV without electrical power supply when the core is in a high pressure state. Coolant for injection is drawn from either the Condensate Storage Tank (CST) or the S/P. Individual piping is provided (steam supply and exhaust line and coolant suction and injection line) as well as the necessary valves, instrumentation and controls.
The configuration of the RCIC is shown on Figure 13.4-2.

(b) HPCF
Two divisions of HPCF are provided. Each division of the HPCF consists of a pump which performs coolant injection into the RPV when the reactor is in high or low pressure state. Coolant for injection is drawn from either the CST or the S/P. Individual piping is provided to each division (coolant suction and injection line) as well as the necessary valves, instrumentation and controls.
The configuration of the HPCF is shown on Figure 13.4-3.

(c) LPFL
The LPFL is a mode of the RHR which can be performed by any one of the three RHR divisions. The RHR in LPFL mode draws coolant from the S/P, cools it by passing it through the RHR heat exchangers, and injects it into the RPV when the reactor is in a low pressure state. The RHR components performing the LPFL are the RHR Pumps (one per division), the RHR Heat Exchangers (two per division), and the individual piping of each division (coolant suction line from the S/P and injection line into the RPV) as well as the necessary valves, instrumentation and controls.
The configuration of the LPFL is shown on Figure 13.4-4.

(d) ADS
The ADS consists of seven out of the sixteen SRVs connected on the Main Steam (MS) lines, which are provided with the ADS function to depressurise the RPV by

relieving high pressure steam to the S/P where it is condensed. The SRVs with ADS function are provided with one dedicated accumulator for ADS operation in addition to the accumulator for relief operation, and three additional solenoid valves dedicated for ADS operation.
The configuration of the SRVs is shown on Figure 13.4-6.

Alternative systems for core cooling configuration

(e) FLSS
The FLSS consists of two trains of two pumps each which perform coolant injection into the RPV when the reactor is in low pressure state as well as flooding of various locations. A dedicated water source (FLSS Water Storage Tanks, ten units) and individual piping (suction lines from the tanks and injection lines to the various destinations) are provided to the FLSS, as well the necessary valves, instrumentation and controls.
The configuration of the FLSS is shown on Figure 13.4-5.

(f) RDCF
The RDCF consists of seven SRVs out of the sixteen connected on the MS lines and which are not provided with the ADS function. As the ADS SRVs, the SRVs with RDCF function depressurise the RPV by relieving high pressure steam to the S/P where it is condensed. The SRVs with the RDCF function are provided with one dedicated accumulator for RDCF operation in addition to the accumulator for relief operation, and two additional solenoid valves that control the RDCF accumulator. In addition, four out of the seven RDCF SRVs can be actuated by dedicated nitrogen gas cylinders through a switching valve for long-term depressurisation events without relying on power supply availability.

(4) Modes of Operation

Operation of the systems that form the ECCS network are summarised as follows:

(a) RCIC
The RCIC injects water into feedwater line (B), using a pump driven by an incorporated steam turbine. The RCIC steam supply line branches off one of the MS lines leaving the RPV and goes to the turbine of the RCIC Pump. The RCIC is designed to supply cooling water into the RPV in the event of transients while the reactor is in high pressure condition, or until the vessel pressure drops to the point at which the LPFL can be placed in operation in the event of a LOCA. The RCIC is initiated automatically upon a predetermined reactor low water level or high drywell pressure signal.

(b) HPCF
The HPCF pumps water through a flooder sparger mounted within the RPV above the reactor core. Coolant is supplied over the entire range of system operation pressures. The primary purpose of the HPCF is to maintain reactor vessel inventory after transients as a backup of the RCIC, or in the event of a small break which does not significantly depressurise the RPV. The HPCF starts operating automatically upon the reactor low water level signal or the drywell high pressure signal in order to maintain the core covered with water.

(c) SRV

As a support for the delivery of high pressure core cooling, the SRVs through the safety valve and relief valve functions that are described in Generic PCSR section 12.3.3.3, release the steam generated due to the decay heat to the S/P during reactor core cooling by the RCIC and HPCF in the event of reactor isolation event or a pipe break outside the PCV, which do not depressurise the reactor. This operation is performed simultaneously with overpressure protection of the Reactor Coolant Pressure Boundary (RCPB).

On the other hand, the ADS utilises a sub-set of the reactor SRVs to reduce reactor pressure in the event that the HPCF and the RCIC cannot maintain cooling or reactor water level so that injection can be performed with a higher flowrate with the HPCF at low pressure and the LPFL. When the RPV pressure is reduced within the injection capacity of the RHR, this system starts up in LPFL mode and provides inventory makeup so that acceptable post-accident temperatures are maintained. Specifically, the piston actuator of the SRVs is automatically driven by the actuation signals to forcibly open the SRVs and quickly reduce the nuclear reactor pressure in the event of a small or a medium piping break inside or outside the PCV. Thus, makeup from the LPFL can be provided into the RPV. Seven valves out of the 16 SRVs are provided with this ADS function.

(d) LPFL

The LPFL has three independent loops and delivers water to the core at relatively low reactor pressures. The primary purpose of the LPFL is to provide inventory makeup and core cooling in the event of a large break that depressurises the reactor, and to provide containment cooling. Following ADS initiation, the LPFL can also provide inventory makeup and core cooling in the event of a smaller break.

(e) FLSS

Operation modes of the FLSS are described in Chapter 16 : Auxiliary Systems section 16.7.3.1. The function as a secondary means for core cooling includes only the RPV injection mode.

(f) RDCF

Operation modes of the RDCF are described in Chapter 16 : Auxiliary Systems section 16.7.3.3.

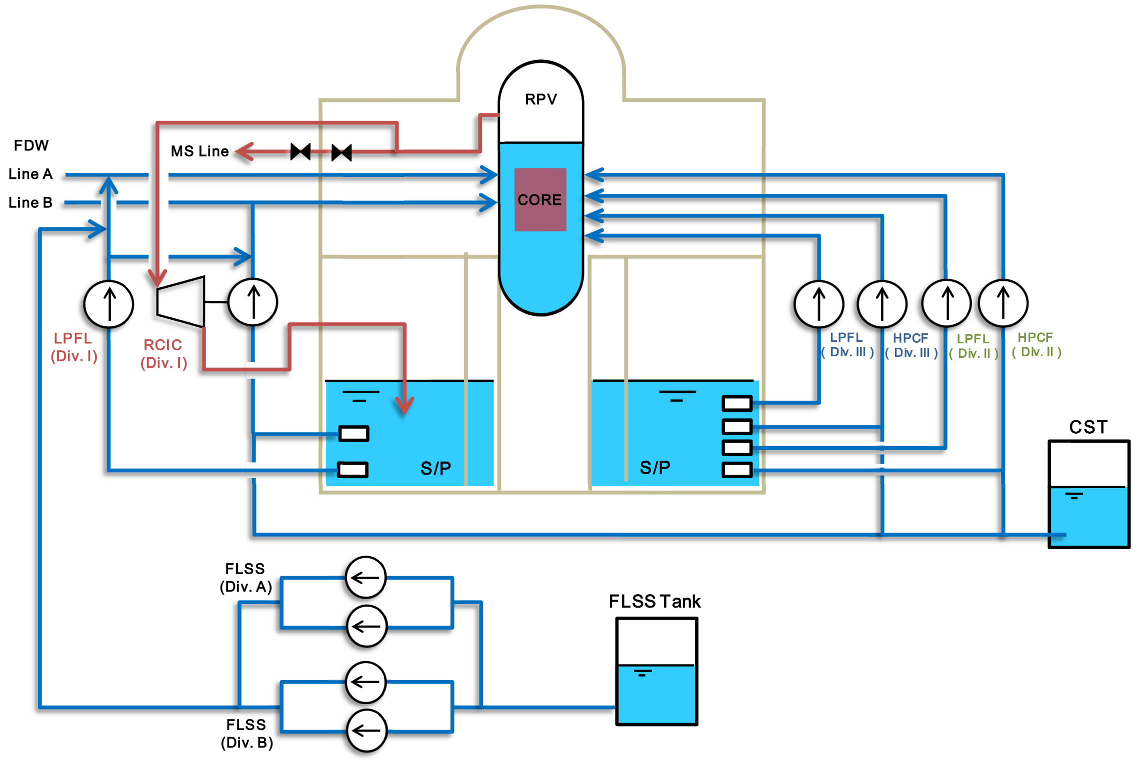

Figure 13.4-1 : Outline of the ECCS and Alternative Systems for Core Cooling Network

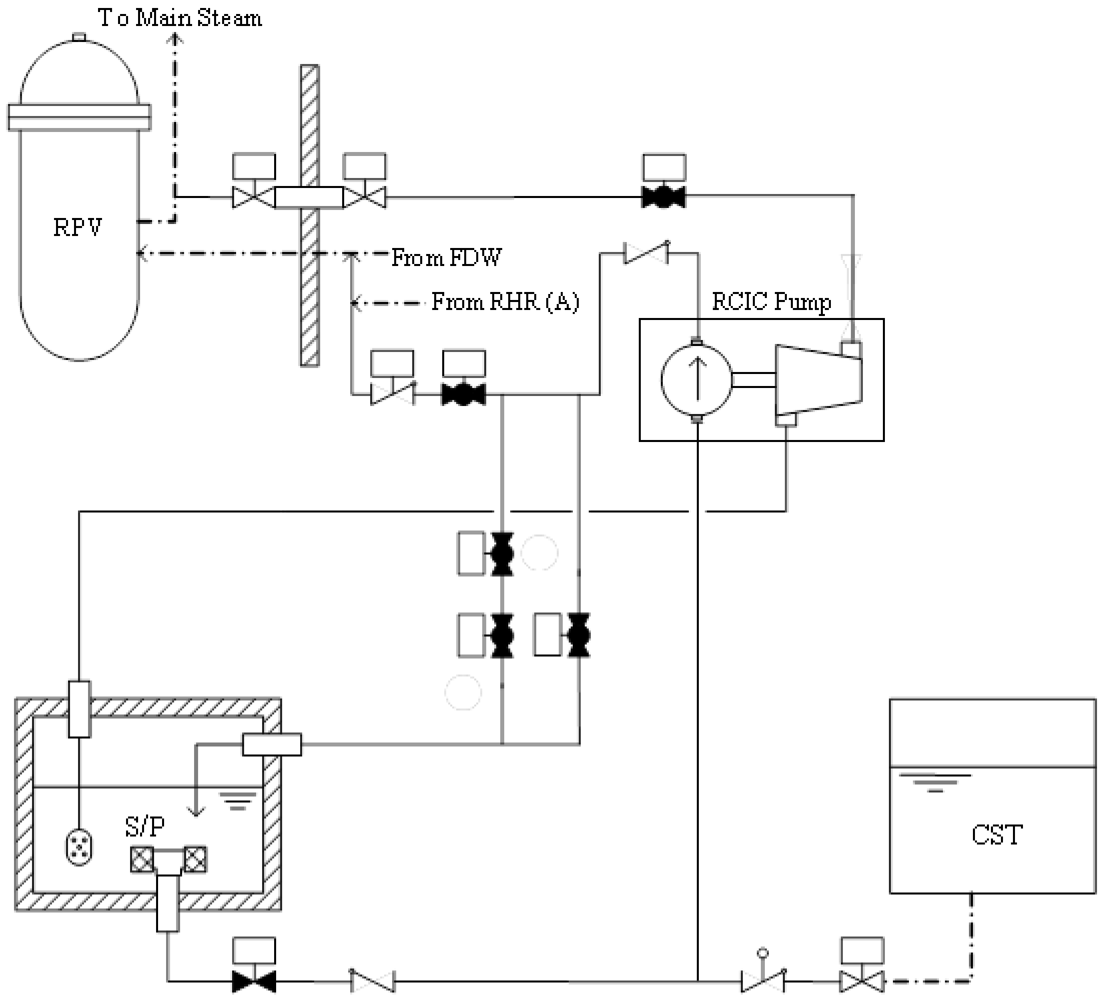

Figure 13.4-2 : Outline of the RCIC

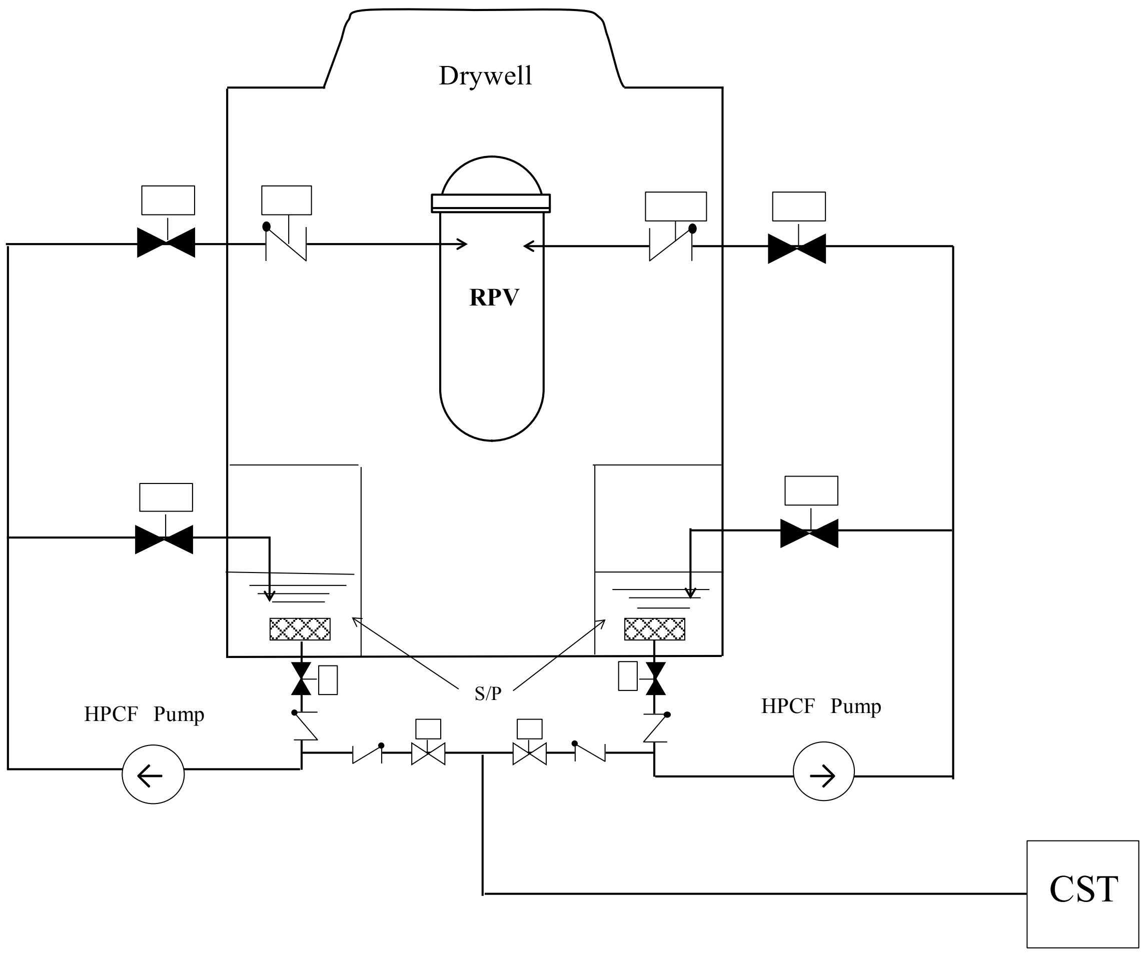

Figure 13.4-3 : Outline of the HPCF

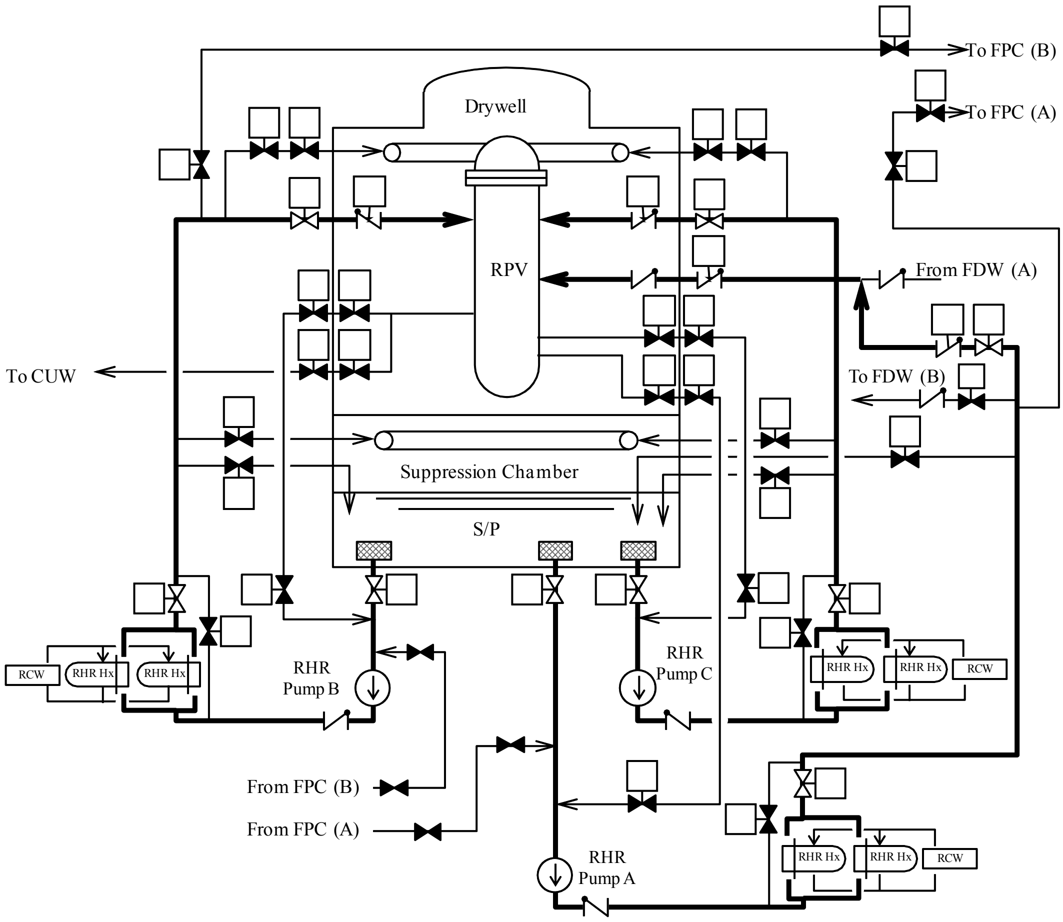

Figure 13.4-4 : Outline of the LPFL mode of the RHR

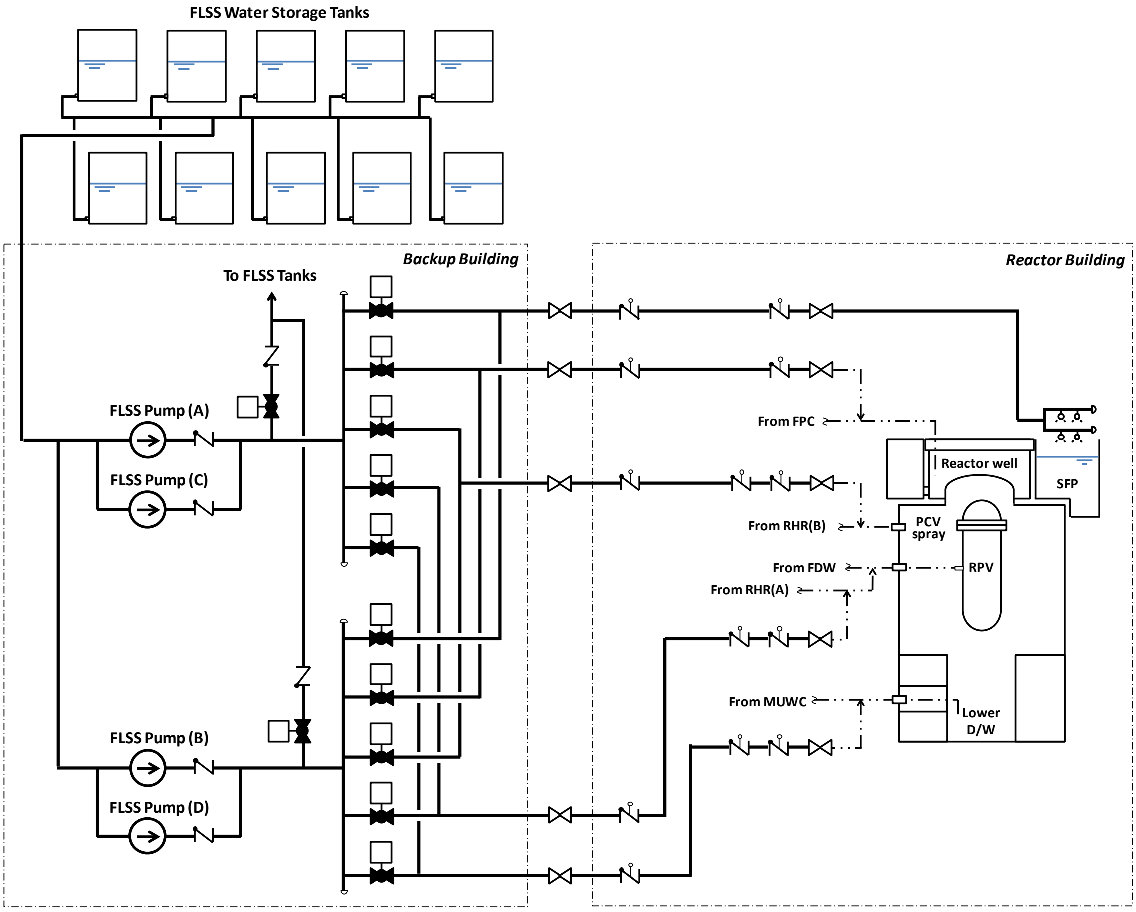

Figure 13.4-5 : Outline of the FLSS

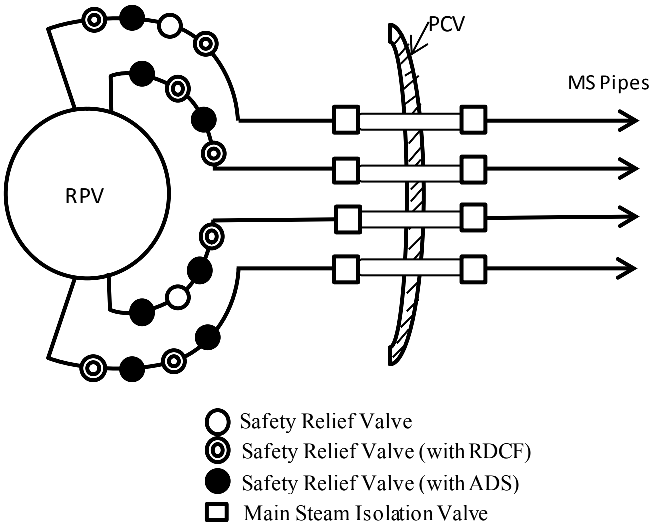

Figure 13.4-6 : Outline of the SRVs

(5) Design Bases

This section describes the design bases for the ECCS and the alternative systems for core cooling.

The ECCS and the alternative systems for core cooling have been designed to meet the following Safety Functional Claims (SFCs). The claims put on the system come from the BSCs on Emergency Core Cooling System [Ref-13.4-1]. The containment functions claimed on the portions included in the PCVB and RCPB for the LPFL and the ADS are cited in Chapter 12 : Reactor Coolant Systems, Reactivity Control Systems and Associated Systems section 12.3.5.4 related to the RHR and 12.3.5.2 related to the NB, respectively.

The linkage between the SFCs of the ECCS with the FSFs and the HLSFs is shown in Appendix A. The FSFs and HLSFs are defined in Chapter 5 : General Design Aspects.

<u>Fault Conditions</u>

[Primary means against design basis faults]

(a) The RCIC is a principal means to provide reactor core cooling as part of the ECCS when the RPV is in high pressure state and in the interval it is being depressurised 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 frequent faults such as loss of the normal feedwater supply and infrequent faults such as LOCA. [RCIC SFC 2-1.1]

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

(b) The HPCF is a principal means to provide reactor core cooling as part of the ECCS when the RPV is in high pressure state and in the interval it is being depressurised 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 frequent faults such as loss of the normal feedwater supply and infrequent faults such as LOCA. [HPCF SFC 2-1.1].

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

(c) The HPCF 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. [HPCF SFC 2-1.2].

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

(d) 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 Category A and the components to deliver it are designed to meet Class 1 requirements.

(e) 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 inside the PCV. [NB SFC 2-1.3].

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

(f) 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 outside the PCV. [NB SFC 2-1.4].

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

(g) The RCIC is capable of providing reactor core cooling during at least 8 hours 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 a design basis fault with loss of Class 1 reactor core cooling SSCs due to CCF. [RCIC SFC 2-1.2]

[Secondary means against design basis faults]

(h) The FLSS is the secondary means to provide reactor core cooling in order to prevent significant damage to the fuel and minimise the reaction between the fuel cladding and the reactor coolant sufficiently in the event of design basis faults with all the primary reactor core cooling means (ECCS) failed. [FLSS SFC 2-2.1]

This function is categorised as Category A and the components to deliver it are designed to meet Class 2 requirements. It is developed in Chapter 16 : Auxiliary Systems section 16.7.3.1 related to the FLSS.

(i) The RDCF is an alternative means to depressurise the RPV in order to provide reactor core cooling in low pressure state with the FLSS in the event of design basis faults where the primary means (ECCS) are not available. [RDCF SFC 2-2.1]

This function is categorised as Category A and the components to deliver it are designed to meet Class 2 requirements. It is developed in Chapter 16 : Auxiliary Systems section 16.7.3.3 related to the RDCF.

(j) The RDCF is the principal means to depressurise the RPV in order to provide reactor core cooling in low pressure state with the FLSS after RCIC operation for the first 24 hours in the event of design basis faults such as SBO or Class 1 CCF. [RDCF SFC 2-2.2]

This function is categorised as Category A and the components to deliver it are designed to meet Class 2 requirements. It is developed in Chapter 16 : Auxiliary Systems section 16.7.3.3 related to the RDCF.

(k) The RDCF with switching valves is the principal means to maintain RPV depressurisation in order to provide reactor core cooling in low pressure state with the FLSS in the event of design basis faults such as SBO or Class 1 CCF after the first 24 hours. [RDCF SFC 2-2.3]

This function is categorised as Category A and the components to deliver it are designed to meet Class 3 requirements. It is developed in Chapter 16 : Auxiliary Systems section 16.7.3.3 related to the RDCF.

[Others]

(l) The RCIC 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. [RCIC SFC 4-7.1].

This function is categorised as Category A and the components to deliver it are designed to meet Class 1 requirements. It is developed in section 13.3.3.2 of this chapter.

(m) The HPCF 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. [HPCF SFC 4-7.1].

This function is categorised as Category A and the components to deliver it are designed to meet Class 1 requirements. It is developed in section 13.3.3.2 of this chapter.

Normal and Fault Conditions

(n) 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].

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 in Generic PCSR section 12.3.3 related to the RCPB.

(o) 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].

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 in Generic PCSR section 12.3.3 related to the RCPB.

Normal Conditions

(p) The RCIC steam supply line contains reactor coolant (steam) up to the turbine stop valve during power operation. 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. [RCIC SFC 4-3.1].

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

(q) The RCIC outside the RCPB contains radioactive material. A breach could lead to a release of radioactive material of dose consequences that are relatively low. [RCIC SFC 4-4.1].

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

(r) The HPCF portion outside the RCPB contains radioactive material. A breach could lead to a release of radioactive material of dose consequences that are relatively low. [HPCF SFC 4-4.1].

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

The Safety Functional Claims related to the containment and confinement functions of the related systems (RHR, FLSS, SRVs) are contained in Chapters 12 and 16 in the sections related to these systems.

System Design Description

This section describes the design of the ECCS and the alternative systems for core cooling to support and justify the delivery of [RCIC SFC 2-1.1], [HPCF SFC 2-1.1], [HPCF SFC 2-1.2], [NB SFC 2-1.3], [NB SFC 2-1.4], [RHR SFC 2-1.1] [RCIC SFC 2-1.2], [FLSS SFC 2-2.1], [RDCF SFC 2-2.1], [RDCF SFC 2-2.2] and [RDCF SFC 2-2.3].

Additional design description can be found in the system specifications ([Ref-13.4-2] [Ref-13.4-5] [Ref-13.4-8] [Ref-13.4-12] [Ref-13.4-14] and [Ref-13.4-17]), and the piping and instrumentation diagrams ([Ref-13.4-3] [Ref-13.4-4] [Ref-13.4-6] [Ref-13.4-7] [Ref-13.4-9] [Ref-13.4-10] [Ref-13.4-11] [Ref-13.4-13] [Ref-13.4-15] [Ref-13.4-16] and [Ref-13.4-18]).

(1) Overall System Design and Operation

(a) RCIC

The RCIC consists of a pump assembly driven by an incorporated steam-driven turbine. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the MS lines (leaving the RPV) within the PCV and goes to the turbine with drain transfer provision to the Main Condenser. The turbine exhausts to the S/P with vacuum breaking protection. Makeup water is supplied from either the CST and the S/P RCIC Pump discharge lines include the main discharge line to the feedwater line, a test return line to the S/P and a minimum flow bypass line to the S/P.

During normal plant operation, the RCIC is on standby with the motor-operated valves in their normal positions (open/closed). The Make-up Water Condensate system (MUWC) is operating to keep the RCIC Pump discharge line filled during this mode to minimise the time lag between a starting signal and initiation of flow into the RPV, and to minimise momentum forces associated with accelerating fluid into an empty pipe.

In the event of a transient, following reactor scram, steam generation in the reactor core continues at a reduced rate due to the core fission product decay heat. The turbine bypass system diverts the steam to the Main Condenser, and the Feedwater System (FDW) supplies the makeup water required to maintain the RPV inventory. In the event that the RPV is isolated (Main Condenser unavailable), or the feedwater supply is unavailable, the SRVs are provided to automatically or passively maintain reactor pressure within desired limits. The water level in the RPV drops due to continued steam generation by decay heat. Upon reaching a predetermined low level (Level 2, auxiliary feedwater mode operating signal) from four independent and redundant sensors (two-out-of-four logic), the RCIC is initiated automatically. The RCIC pump supplies water from the CST or the S/P to the RPV, the CST being the source by default. In the event CST water level falls below a predetermined setpoint, the pump suction automatically transfers from the CST to the S/P. The RCIC turbine is driven with decay heat steam from the RPV, and exhausts the steam to the S/P. The RCIC also actuates on LOCA signal (low reactor water level (Level 1.5) or high drywell pressure signal (13.7kPa [gauge]) from four independent and redundant sensors (two-out-of-four logic), to maintain the water level in the RPV in the event of a small break until vessel pressure drops to a point at which the LPFL can be placed in operation. In that case, water source switch to the S/P is performed also if the S/P water level rises above a predetermined setpoint.

The valve on the steam supply line shuts off automatically upon receipt of a high reactor water level signal (Level 8), thereby stopping RCIC pump operation. The valves reopen to restart RCIC operation if the reactor water level decreases to a low level again (Level 2). The RCIC is capable of manual initiation by the operator in case of failure of automatic initiation or shutoff. The RCIC continues operation until depressurization of the RPV is performed in order to start operation of the low-pressure injection cooling systems.

During RCIC operation, the S/P acts as the heat sink for steam generated by reactor decay heat.

The RCIC is capable of operating without supply of electric power, except for the motor operated valves and the instrumentation and controls, which is provided by DC batteries. Therefore, even in the case of loss of offsite power (LOOP) and loss of all the emergency AC power sources (station blackout), the RCIC can provide the makeup water required to maintain the RPV inventory. DC batteries and CST are designed to allow operating the RCIC during 8 hours, after which operation to provide makeup water to the reactor is handed over to another system (such as the FLSS) if power supply is still not recovered.

(b) HPCF

The HPCF starts operating automatically upon detection of reactor low water level (Level 1.5) or drywell high pressure (13.7kPa [gauge]) from four independent and redundant sensors (two-out-of-four logic). The HPCF is capable of manual initiation as well if required. Suction is taken from the water source with the pump discharge diverted through the minimum flow lines to the S/P until sufficient pump flow rate and pressure have been developed to discharge into the RPV. The HPCF Pump and injection valve are designed to actuate and allow water injection into the reactor at rated flowrate within the time required by the safety analysis presented in Chapter 24 : Design Basis Analysis after receipt of initiation signal.

The injection valves on the discharge lines automatically close after automatic initiation if the RPV water level increases to a determined high level (Level 8). The valves reopen if the reactor water level decreases to a low level again (Level 1.5). Each minimum flow valve opens automatically if the main line flow rate is low, allowing the HPCF Pump to discharge at minimum flow rate into the S/P. Once the HPCF emergency mode is initiated, the system remains operating until it is manually stopped by the operator.

The HPCF takes suction from the either the CST or the S/P, with the CST being the default source. In the event CST water level falls below a predetermined setpoint or S/P water level rises above a predetermined setpoint, the pump suction automatically transfers from the CST to the S/P. The HPCF suction lines are independent from the RHR.

(c) ADS

If the RCIC and HPCF cannot maintain the reactor water level, the ADS reduces the reactor pressure so that flow from the RHR operating in the LPFL mode enters the RPV in time to cool the core and maintain fuel cladding temperature below acceptable limits.

The ADS employs seven out of the 16 SRVs which are provided with the ADS function of the NB to relieve high pressure steam directly to the S/P and condense it. Each of the SRVs with ADS function is provided with an additional accumulator dedicated for ADS operation in addition to the accumulator for SRV relief operation. In the same way as the accumulator for relief operation, the ADS accumulator of the SRVs with ADS function is connected to the High Pressure Nitrogen Gas Supply System (HPIN) and the nitrogen gas cylinder rack for backup as additional supply. The SRVs are designed so they can open by the nitrogen gas stored in the accumulators even if the nitrogen gas supply system in the PCV is damaged. A check valve is mounted on the nitrogen supply line to each accumulator so that the internal pressure of the accumulator does not drop rapidly in the event that the supply system on the upstream side of the accumulator is damaged. SRVs with the ADS function are provided with four solenoid valves to control them from the MCR, one for operation of the accumulator for SRV relief function and three for operation of the accumulator for ADS function. The solenoid valves are remotely and independently operated.

The SRVs with the ADS function are actuated automatically upon detection of a LOCA and maintained opened by the ADS accumulator to depressurise the RPV even if the pressure drops below the closing set pressure of the valves. In the event of a LOCA inside the PCV, the actuation signal is initiated by the simultaneous high drywell pressure and low reactor water level (Level 1) signals in conjunction with the signal that one RHR or HPCF Pump is in operation. In the event of a LOCA outside the PCV, the high drywell pressure signal is not immediately generated and the reactor water level continues to decrease if high pressure injection to the RPV is not performed. If the water level reaches the low reactor water level (Level 1), the transient ADS is actuated in conjunction with the signal that one RHR Pump is in operation. The SRVs are designed to open with a sufficient delay after receiving an accident signal to prevent unnecessary operation by a spurious signal since the actuation of this function involves a loss of reactor water to the S/P. In addition to these signals, low neutron flux signal from the Start-up Range Neutron Monitor (SRNM) is applied for ADS actuation permission signal in order to inhibit ADS operation during reactivity control by the Standby Liquid Control system (SLC).

(d) LPFL

The LPFL is composed of three electrical and mechanically independent and separated RHR divisions designated A, B, and C. Each division contains the necessary piping, pumps, valves, heat exchangers, instrumentation and controllers.

The LPFL is an operation mode of the RHR, which by switching the position of the valves, can operate to deliver the low pressure core cooling function as described below. Common design aspects with the rest of the RHR modes are described in Generic PCSR section 12.3.5.4.

The RHR is on standby and the motor-operated valves are at their normal positions (close/open) during plant normal operation. The pump discharge lines are continuously kept filled with condensate water via the MUWC during standby to minimise the time lag between a LPFL starting signal and initiation of flow into the RPV, and to minimise momentum forces associated with accelerating fluid into an empty pipe.

The LPFL supplies sufficient coolant to maintain the fuel cladding temperature below the design basis criteria and to remove the core decay heat during LOCA. During this mode, the RHR draws water from the S/P and injects the water into the RPV outside the core shroud (RHR division A injects water via the feedwater line A and divisions B and C via their respective low pressure lines into the RPV).

The LPFL is initiated automatically upon high drywell pressure or reactor low water level (Level 1) starting signals from four independent and redundant sensors (two-out-of-four logic), and therefore no operator action is required during the first 30 minutes following an accident. The RHR Pump and injection valve are designed to actuate and allow water injection into the reactor at rated flowrate within the time required by the safety analysis in Chapter 24 : Design Basis Analysis after receiving the automatic initiation and reactor low pressure permissive signals. This is the time required for the RHR pump to reach rated revolutions and the injection valve to fully open.

In addition, in the occurrence of a LOCA caused by a LPFL pipe break inside the PCV, the HPCF (B) and (C) can still perform low-pressure water injection into the RPV but injection by the RCIC is stopped. Therefore, for division A of the ECCS, if a LPFL (A) line break occurs inside the PCV, the LPFL (A) injection line is automatically switched to the feedwater line (B) so that low-pressure injection into the RPV can be performed by the RHR (A) Pump.

The LPFL mode can also be initiated manually. Once the LPFL starts, the operation continues until the mode is manually shut off by the operator.

The RHR is provided with a minimum flow bypass line to return the water to the S/P to prevent pump damage until the pump reaches the necessary injection flow rate and pressure while the injection valve is closed. A motor-operated valve on the bypass line automatically closes when the flow rate in the main discharge line is sufficient to provide reactor core cooling.

The RHR is provided with a test line which discharges to the S/P to allow testing of pump function.

(e) FLSS and RDCF

In the event of design basis faults combined with failure of the ECCS to perform its functions of core cooling, the function to provide makeup water to maintain reactor inventory is fulfilled by the FLSS and the RDCF. The FLSS and the RDCF actuate automatically when reactor water level decreases to Level 1. The RDCF triggers reactor depressurisation by sending the opening signal to the seven RDCF SRVs with the additional conditions that depressurisation was not already performed and that at least one FLSS Pump is running. A timer delays the signal to ensure depressurisation is not performed if the ECCS succeeded to recover the water level.

While the reactor is not yet depressurised, the FLSS circulates water from the FLSS Water Storage Tanks through the minimum flow line, which returns water to the tanks. The FLSS RPV Injection Valves open automatically when RPV pressure has dropped sufficiently to allow injection into the RPV by the FLSS Pumps. The FLSS Pumps then inject water from the FLSS Water Storage Tanks to the RPV, through the feedwater line (A) and reactor water level is recovered. When water level reaches Level 8, an interlock switches the FLSS to standby, the pumps keep running but circulate water through the minimum flow line. If water level drops again to Level 3, RPV injection is resumed automatically, such that the reactor water level is controlled between Level 3 and Level 8.

The FLSS is designed to have the capability to keep operating during seven days.

Further details regarding FLSS and RDCF operation are provided in Chapter 16 : Auxiliary Systems sections 16.7.3.1 and 16.7.3.3, respectively.

(2) Equipment Design and Operation

(a) RCIC Pump

(i) Purpose
The purpose of the RCIC pump is to provide water from the S/P or the CST to inject into the reactor for core cooling. This is related to [RCIC SFC 2-1.1] and [RCIC SFC 2-1.2].

(ii) Configuration and Operation
The RCIC Pump incorporates a steam-driven turbine which drives the pump assembly to deliver high pressure core flooding for reactor core cooling. This flow rate satisfies the required minimum flow rate to maintain reactor water level according to the safety analysis in Chapter 24 : Design Basis Analysis. Therefore, one pump delivering approximately 182m^3/h is provided.

(iii) Performance
The RCIC Pump is designed to perform as follows to ensure the delivery of the safety functions:

Table 13.4-1 : RCIC Pump Performance Parameters

Item	Value
Capacity (m^3/h)	approx. 182
Total Head (m)	approx. 900 to 186

The RCIC Pump is designed such that it can be initiated with the discharge valves closed and reach rated flow within the time required by the safety analysis in Chapter 24 : Design Basis Analysis after receiving the initiation signal.

(b) HPCF Pump

(i) Purpose
The purpose of the HPCF pump is to provide water from the S/P or the CST to inject into the reactor for core cooling in order to deliver [HPCF SFC 2-1.1] and [HPCF SFC 2-1.2].

(ii) Configuration and Operation
Each division of the HPCF is provided with one turbo type pump of approximately 182m^3/h design flow rate driven by an induction motor to deliver core flooding for reactor core cooling over the whole range of operating pressures. This flow rate satisfies the required minimum flow rate to maintain reactor water level according to the safety analysis in Chapter 24 : Design Basis Analysis. Therefore, a total of two HPCF pumps delivering approximately 182m^3/h at high RPV pressure and 727m^3/h at low RPV pressure each are provided.

(iii) Performance
The HPCF Pump is designed to perform as follows to ensure the delivery of the safety functions:

Table 13.4-2 : HPCF Pump Performance Parameters

Item	Value	
	High Pressure	**Low Pressure**
Capacity (m^3/h)	approx. 182	approx.727
Total Head (m)	approx.890	approx.190

The HPCF 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 Chapter 24 : Design Basis Analysis after receiving the initiation signal.

(c) RHR Pump

The design and operation of the RHR pump in order to deliver [RHR SFC 2-1.1] is described in Chapter 12 : Reactor Coolant Systems, Reactivity Control Systems and Associated Systems section 12.3.5.4 about the RHR.

(d) SRVs with ADS Function

The SRVs with the ADS function are activated in the event of a LOCA to depressurise the reactor core in order to allow activation of the LPFL mode of the RHR for core cooling in addition to the HPCF and the RCIC in order to deliver [NB SFC 2-1.3] and [NB SFC 2-1.4]. The design and operation of the SRVs with ADS function is described in Chapter 12 : Reactor Coolant Systems, Reactivity Control Systems and Associated Systems section 12.3.5.2 about the NB.

(e) FLSS Pump

The design and operation of the FLSS pump in order to deliver [FLSS SFC 2-2.1] is described in Chapter 16 : Auxiliary Systems section 16.7.3.1 about the FLSS.

(f) SRVs controlled by the RDCF

The design and operation of the SRVs controlled by the RDCF in order to deliver [RDCF SFC 2-2.1], [RDCF SFC 2-2.2] and [RDCF SFC 2-2.3] is described in Chapter 12 : Nuclear Boiler System section 12.3.5.2 about NB.

(3) Main Support Systems

(a) Instrumentation and Control System

The system supporting the ECCS with instrumentation and control is the Safety Class 1 Safety System Logic and Control System (SSLC) and the system supporting the alternative systems for core cooling is the Safety Class 2 Hardwired Backup System (HWBS). The design and the claims on the SSLC are addressed in Chapter 14 : Control and Instrumentation.

(i) Instrumentation
Instrumentation is provided to measure and monitor the operating conditions of the ECCS and the alternative systems for core cooling components necessary for the delivery of reactor core cooling. The main provisions for instrumentation are described as follows:

- ECCS 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.
- Alternative means for core cooling (FLSS, RDCF) can be controlled and monitored from either the MCR or the B/B control room.
- Pressure detectors are provided in the pump discharge lines in order to verify the pump seal water conditions and to raise an alarm if low pressure is detected. They also control opening and closing of the minimum flow lines.
- For the RCIC, pressure detectors are provided at the turbine inlet and outlet on the steam line.
- Local pressure indicators are installed at the pump suction lines to monitor the pump Net Positive Suction Head (NPSH) and head.

- Flow-meters are mounted on each injection system injection lines to monitor injection flow rate.
- Instruments to detect the water level in the CST and S/P are provided to control RCIC and HPCF suction valves switches.

(ii) Control
The main control provisions related to the delivery of the reactor core cooling are summarised as follows:

ECCS Actuation parameters
- Reactor Water Level 8: the steam supply for RCIC pump is stopped and the HPCF injection valve closed if RCIC or HPCF divisions are operating.
- Reactor Water Level 2: initiation of RCIC water supply at this level, which is set at a sufficiently low level to prevent RCIC spurious initiation if the feedwater supply is still available after reactor scram at Level 3 (auxiliary feedwater mode).
- Reactor Water Level 1.5: initiation of RCIC (LOCA mode) and HPCF.
- Reactor Water Level 1: initiation of LPFL mode of the RHR and Transient ADS. ADS is initiated under the additional condition of high drywell pressure.
- High drywell pressure (LOCA inside containment): actuation of RCIC (LOCA mode), HPCF and LPFL. ADS is initiated under the additional condition of Reactor Water Level 1.

HPCF
- The reactor coolant source is automatically switched to the S/P upon low CST level or high S/P level.

RCIC
- The reactor coolant source is automatically switched to the S/P upon low CST level. If the RCIC actuation signal in LOCA mode (Level 1.5 or high drywell pressure) was sent, the coolant source is also switched upon high S/P level.

LPFL
- The RHR Pumps start automatically after confirmation of power supply once the initiation signal has been received.
- The injection valve receives the opening signal upon system initiation but is maintained closed until RPV pressure stays high. The pumps circulate S/P water through the minimum flow lines while the RPV is still not depressurised.
- Pressure transmitters mounted on the RHR Pump discharge lines detect that at least one RHR Pump is operating to transmit the permission signal for ADS operation.
- Pressure difference between the two FDW pipes inside the PCV is monitored to detect a pipe break of Feedwater line (A) (LOCA) and trigger switch of LPFL (A) injection line from Feedwater line (A) to line (B).

FLSS, RDCF
- The FLSS and the RDCF are automatically actuated when reactor water level decreases to Level 1, with a timer provided so that the alternative systems are initiated only if the primary systems (ECCS) have failed.

- The injection valve receives the opening signal upon system initiation but is maintained closed until RPV pressure stays high. The pumps circulate tanks water through the minimum flow line while the RPV has not been depressurised.
- A pressure transmitter mounted on the FLSS Pumps discharge line detect that FLSS Pumps are operating to transmit the permission signal for RDCF operation.
- Once FLSS starts injecting into the reactor, the injection valve opens and closes when reactor water level hits Level 8 and 3 respectively, so that the FLSS maintains the reactor water between those two levels.

(b) Power Supply System

Power supply for components, valves, instrumentation and controllers come from the Safety Class 1 Electrical Power Distribution System (EPS) for the ECCS and from the B/B Class 2 EPS for the alternative systems for core cooling. The detailed design and the claims on the EPS are addressed in Chapter 15 : Electrical Power Supplies.

(i) The ECCS is connected to separated and independent divisions of the Safety Class 1 AC and DC EPS supplying the required power to all electrical components in each division (ECCS Division I, II and III are connected to Safety Class 1 EPS Division I, II and III, respectively). The normal AC and DC power supply to the ECCS electrical components, valves, instrumentation and controllers is provided by the external grid. In addition, the Safety Class 1 Emergency Diesel Generators of Division I, II and III provide power for all ECCS components in Division I, II and III, respectively, and the ADS in the event of Loss of Off-site Power (LOOP).

(ii) The RDCF and the FLSS are connected to physically separated and independent divisions of the B/B Class 2 AC and DC EPS, supplying the required power to all electrical components in each division (RDCF solenoid valves are connected to B/B class 2 EPS divisions A and B, and the FLSS Division A and B are connected B/B Class 2 EPS system 1 and 2, respectively). The normal AC and DC power supply to the FLSS and RDCF electrical components, valves, instrumentation and controllers is provided by the external grid. In addition, the Safety Class 2 B/B Generators 1 and 2 provide power for FLSS division A and B components respectively, and the RDCF in the event of LOOP or Station Blackout (SBO).

(iii) In the event of loss of all AC power supply (LOOP and failure of AC emergency power sources), the batteries of the Safety Class 1 DC power supply system Division I provide electrical power to the RCIC valves, instrumentation and controllers. The RCIC Pump is driven by steam and does not need electrical power to operate.

(c) Reactor Building Cooling Water System (RCW)

Cooling water for ECCS components is supplied by the Safety Class 1 RCW. The detailed design and the claims on the RCW are described in Chapter 16 : Auxiliary Systems. Cooling water supply is not required for the alternative systems for core cooling.

(i) The RCW supplies water to the HPCF Pump mechanical seal and the motor bearing coolers, the RHR Heat Exchangers, RHR Pumps, motors, bearings and seal water cooling equipment.
(ii) The RHR and HPCF are connected to independent and physically separated RCW divisions. RHR division A components are supplied cooling water by RCW division A, RHR division B and HPCF division B components are supplied cooling water by RCW division B and RHR division C and HPCF division C components are supplied cooling water by RCW division C.

(4) System Architecture

(a) Redundancy

The ECCS consists of three redundant divisions I, II, and III, each one of them being provided with high pressure and low pressure core cooling systems and with their respective pumps, heat exchangers, strainers, piping, valves, test line, minimum flow line and instrumentation. In addition, the ADS working in conjunction with all divisions is provided with redundant SRVs, accumulators and solenoid pilot valves. Furthermore, the systems supporting ECCS functions such as the C&I, power supply, cooling water and HVAC also have redundant divisions, dedicated for the corresponding ECCS division. The configuration is as shown below:

(i) Division I: RCIC + LPFL (A)
(ii) Division II: HPCF (B) + LPFL (B)
(iii) Division III: HPCF (C) + LPFL (C)
(iv) All divisions: ADS

Similarly, the FLSS consists of two trains (divisions A and B) with dynamic components such as pumps, valves and instrumentation being redundant and individual to each train. The RDCF is working in conjunction with both divisions and is provided with redundant SRVs, accumulators and solenoid pilot valves. Furthermore, the systems supporting FLSS and the RDCF functions such as the C&I, power supply and HVAC also have redundant divisions, dedicated for the corresponding FLSS division.

Therefore, for both the ECCS and the alternative means for core cooling, the configuration is such that, single failure of any dynamic mechanical or electrical component under the worst permissible availability state does not prevent the delivery of the safety function.

(b) Independence

The three divisions forming the ECCS are functionally independent and physically separated in different locations within the Reactor Building (R/B) to prevent failure of a component in one of the divisions from leading to a common cause failure of all divisions. The components forming the ADS are independent and are separated as far as practicable within the PCV to prevent failure of a single component affecting others or leading to common cause failure. Furthermore, the redundant supporting systems for each division of ECCS and ADS (C&I, power supply, cooling water and HVAC) are independent and separated as well.

Similarly, the two trains of the FLSS are functionally independent to prevent failure of a component in one train from leading to a common cause failure of both divisions; and physically separated from lower safety class items of which the failure could result in failure of the FLSS. The RDCF components are independent and separated in the same way ADS components are. Furthermore, the redundant supporting systems for each division of FLSS and the RDCF (C&I, power supply and HVAC) are independent and separated as well.

(c) Diversity

The design of the ECCS and the alternative systems for core cooling (FLSS, RDCF) has been carried out taking into account diversity in terms of structure and components, operating conditions and functioning principles so that the risk of an event leading to a common failure of both the primary and the secondary means for core cooling is reduced. They are also physically separated to prevent an internal hazard from leading to their common failure. The same criteria are applied to the support systems (C&I, power supply and HVAC) so that failure of one of them does not result in full unavailability of the core cooling function.

Assumptions, Limits and Conditions for Operation

In order to ensure that the ECCS and the alternative systems for core cooling are operated within safety limits and the design requirements from the safety case are met during the operating regime, appropriate LCOs, 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 the Basis on Safety Cases on Emergency Core Cooling System ([Ref-13.4-3]).

- Six ECCS RPV injection subsystems and seven ADS valves shall be operable during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.
- Two ECCS RPV injection subsystems shall be operable during cold shutdown and refuelling for the delivery of the SFCs claimed when required.
- Two FLSS trains shall be operable during power operation, start-up, hot shutdown, cold shutdown and refuelling for the delivery of the SFCs claimed when required.
- Seven RDCF valves shall be operable during power operation, start-up and hot shutdown for the delivery of the SFCs claimed when required.

13.4.2 ECCS System Design Evaluation

Adequate performance of the ECCS is substantiated by evaluating the system response to a loss of reactor coolant due to an instantaneous break of a pipe (LOCA). The analyses included in this subsection demonstrate the UK ABWR ECCS performance for the entire spectrum of postulated pipe break sizes that are considered within the design basis.

(1) Events analysed

The key events evaluated assume that reactor coolant flows out of the reactor pressure vessel via the break, due to damage to piping which is part of the RCPB and associated equipment during rated power operation of the reactor. Consequently the cooling capacity of the reactor core is reduced.

Pipe breaks of the MS line, FDW line, LPFL line, HPCF line, RHR line, and Bottom Drain line which connect to the RPV are analysed and evaluated. Note that a breach of components within the RCPB that are designated as VHI components, as specified in GDA PCSR Table 12.3-3, are excluded from this evaluation, because such breaches are outside of the design basis, as explained in Chapter 8 : Structural Integrity.

The calculated Peak Cladding Temperature (PCT) for all design basis piping break events in the UK ABWR is almost the same, and the inventory of the coolant reduces the most during the HPCF piping break event. Therefore, the HPCF piping break is selected as the representative break and evaluated below.

(2) Analysis Method

The analysis codes used in the HPCF pipe break analysis are presented in Chapter 24 : Design Basis Analysis subsection 24.5.3.

(3) Acceptance Criteria, Analysis Conditions, and Results

The acceptance criteria applicable to this event, the analysis conditions assumed, and the analysis results are presented in Chapter 24 : Design Basis Analysis subsection 24.8.2.1.

(4) Conclusions

The analysis results show that all of the relevant acceptance criteria are met. Thus adequate performance of the ECCS is substantiated for the UK ABWR.

13.5 Assumptions, Limits and Conditions for Operation

13.5.1 Purpose

This section considers the LCOs that apply specifically to the ESFs 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 BSCs or TR of the corresponding systems:

Reactor Containment Systems and associated systems –

- Primary Containment Vessel (PCV) [Ref 13.3-18]
- Primary Containment Isolation System (PCIS) [Ref 13.3-1]
- Primary Containment Vessel Gas Control Systems [Ref 13.3-2]
- Containment Heat Removal Systems [Ref 13.3-3]
- Reactor Building (R/B) [Ref 13.3-4]
- Standby Gas Treatment System (SGTS) [Ref 13.3-4]

Emergency Core Cooling System and alternative systems for core cooling [Ref 13.4-1]

13.5.2 LCOs Specified for Engineered Safety Features

The LCOs that apply to the ESFs are identified under each individual system within the chapter.

13.5.3 Assumptions for Engineered Safety Features

The Assumptions that apply to the ESFs are identified under each individual system within the chapter.

13.6 Summary of ALARP Justification

This section presents a high level overview of how the ALARP principle has been applied for ESFs, and how this contributes to the overall ALARP argument for the UK ABWR.

Chapter 28 : ALARP Evaluation presents the high level approach taken for demonstrating ALARP across all aspects of the design and operation. It presents an overview of how the UK ABWR design has evolved, the further options that have been considered across all technical areas resulting in a number of design changes, and how these contribute to the overall ALARP case. The approach to undertaking ALARP Assessment during GDA is described in the GDA ALARP Methodology [Ref 13.6-1] and SCDM [Ref-13.6-2].

For the mechanical systems which make up this chapter, this ALARP methodology has been embedded within the design process [Ref 13.6-11]. 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 OPEX, gap analysis, risk assessment, optioneering and design review throughout the design process.

The most significant nuclear safety risks associated specifically with the ESFs of the UK ABWR are:

Containment Systems:

- Inability to adequately control the pressure and temperature increases of the structures that make up the primary containment boundary caused by accidents that result in discharge of reactor coolant into the PCV. Excessive pressures or temperatures could potentially lead to a breach of the PCV, and consequent uncontrolled release of radioactive material.

- Inability to adequately isolate the PCV penetrations in fault conditions that result in discharge of reactor coolant into the PCV. This could potentially lead to unacceptable radioactive release from the PCV via the penetrations.

- Hydrogen gas combustion within the primary or secondary containment.

- Insufficient structural strength of the primary containment to withstand dynamic loads generated during Loss of Coolant Accidents (LOCAs) or during faults that lead to opening of the SRVs.

- Insufficient limitation of discharges of radioactivity to the environment following accidents that release radioactive material into the primary or secondary containment.

- Unacceptably high dose rates associated with normal operation and maintenance of the Containment Systems SSCs.

Emergency Core Cooling System (ECCS):

- Breach of the ECCS components within the Reactor Coolant Pressure Boundary, allowing leakage of radioactive reactor coolant into either the primary or secondary containment.

- Inability to provide sufficient short term or long term cooling of the core in fault or accident conditions, due to low core flow rate.

- Inability to maintain sufficient core inventory in accident conditions that result in a significant loss of core coolant inventory. This could potentially lead to partial core uncovery with consequent overheating of the fuel and embrittlement of the cladding.

- Unacceptably high dose rates associated with normal operation and maintenance of the ECCS SSCs.

The safety of the Japanese ABWR reference design of the ESFs described in this chapter is well understood, using proven technology. Hence a significant aspect of demonstrating the application of relevant good practice in the design of the ESFs of the UK ABWR is to generally adopt the Japanese reference design, with only limited modifications to enhance safety margins even further where it is reasonably practicable to do so. Thus, the UK ABWR design for the ESFs is mainly the same as for the four operating ABWRs in Japan, which together have provided many years of operating experience. Some limited design evolution that was introduced into the latest Japanese ABWR designs that are not yet operating has been incorporated into the reference design. These design evolutions are based on the earlier Japanese designs for which operating experience has been assessed, and also take account of relevant worldwide operating experience.

An example of a significant design change relevant to this chapter that originated from the evolution of the Japanese ABWR, and which has been adopted in the UK ABWR design, is the change from 8 small capacity SRVs with the ADS function on the KK-7 ABWR to 7 large capacity SRVs with the ADS function, as was adopted for the NS-3 ABWR ([Ref-13.6-9], Table 5.3.4-1).

This implementation of relevant good practice 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 UK ABWR design meets the ALARP principle. This has been achieved as described below for the ESF SSCs in the scope of this chapter.

The UK ABWR design reference for ESFs is based on the Japanese reference design with some additional modifications that have been introduced during GDA. The safety case for the UK ABWR design reference of each ESF system is justified in a BSC document (See Reference list in Section 13.8). Each of these BSCs describes an ALARP evaluation that has been performed for the concept reference design of the system following a standard ALARP check list within the general design process approach for Mechanical Engineering (ME) SSCs in the appendix of [Ref-13.6-11]. The ME ALARP verification process in appendix of [Ref-13.6-1] is consistent with the general GDA ALARP methodology in [Ref-13.6-1], including consideration of relevant good practice, identification of potential risk reduction options, consideration of which options are reasonably practicable, and risk assessment.

Details of application of the standard ME design process to specific systems are given in each of the BSCs that support this chapter, for example the BSC for the Containment Isolation System [Ref-13.6-10]. The risk assessments relevant to this chapter, which are part of that process, have taken account of all of the specific risks listed above.

Due to the high level of maturity of the Japanese reference design of the ESFs, there have been few design changes to the particular systems within this Chapter scope as a result of any ALARP assessments performed in GDA. The main exceptions to this are the following:

- Changes to ECCS - "N+2" criteria compliance

  The safety system configuration for the standard ABWR reference design doesn't fully meet the UK good practice when judged against the UK application of the single failure criterion. The interpretation of the single failure criterion is that one division is assumed to be in a test/maintenance state, while the other division is assumed to suffer a single failure occurrence; this is referred to as the 'N+2' criterion. For example on the emergency core cooling system (ECCS) if Division B and C are not operational (i.e., unavailable because of failure or maintenance of two of the three EDGs), leaving only Division A which has RCIC and LPFL/RHR(A) the steam turbine driven RCIC cannot be credited for long-term LOCA makeup when a LOCA break occurs in LPFL/RHR(A) (i.e. in the feedwater line that RHR(A) interfaces with for RPV injection). Therefore, two design changes have been implemented to enable the UK ABWR to accomplish full "N+2" criteria compliance:

  - Introduction of a tie line between LPFL(A) and RCIC with corresponding valves and C&I to maintain the N+2 architecture in the case of a break of the FW line to which LPFL(A) is connected.

  - Replacement of the 50 percent capacity heat exchangers in the RHR system with 100 percent heat exchangers (see Figure 13.6-1, red circles indicate 100 percent capability).

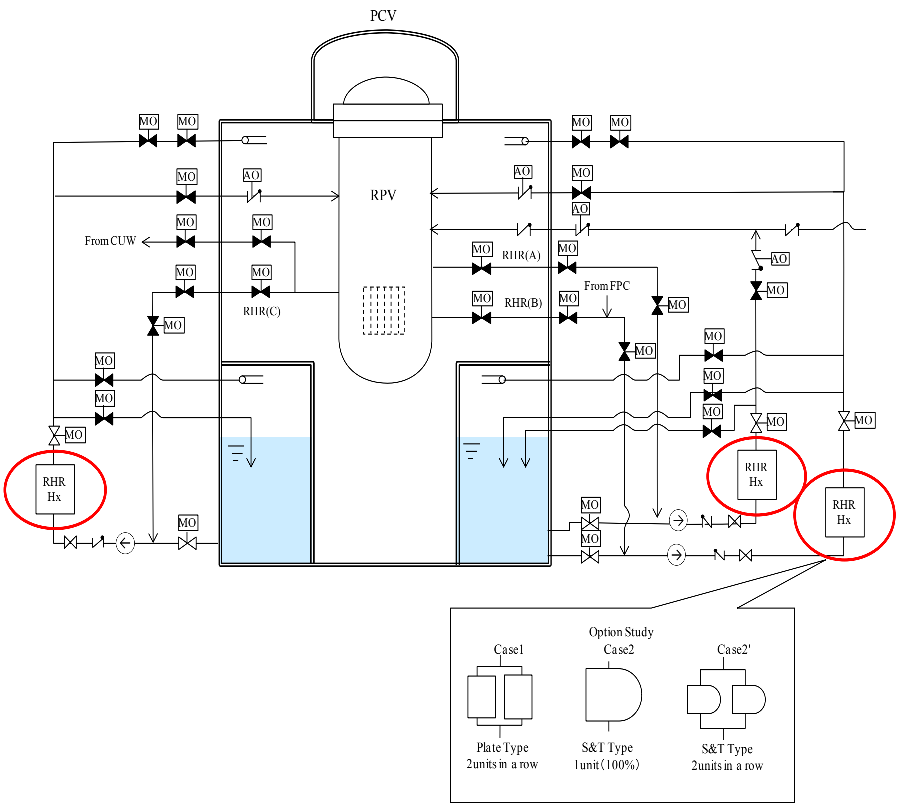

Figure 13.6-1 RHR Change from 50 percent to 100 percent heat exchanger capability

- Countermeasure for the flammable gas control in the PCV
  In the Japanese reference ABWR, thermal recombiners (active system) are installed as the Flammability Control System (FCS). On the other hand, UK ABWR has adopted the Passive Autocatalytic Recombiners (PARs) as a replacement design for the FCS, as described in section 13.3.3.3 of this chapter. An analysis of options showed that the replacement FCS design with PARs had significant advantages over other methods of removing hydrogen such as the active (thermal recombiner type) FCS. PARs require no electrical power, no operator action and no fission products are released to the environment. The relative merits of an active FCS compared with PARs are discussed in [Ref-13.6-5].

- Changes to the RCIC Pump / Turbine
  Improvements to the design of the RCIC system steam driven turbine pump have been implemented. The original design, which consisted of a pump and turbine assembly, has been changed to a single component integrating a pump and a steam-driven turbine as the prime mover for the RCIC. It has a number of advantages:
    - It requires less DC electrical power;
    - It does not require auxiliary functions such as lubricant oil cooling;
    - It is more compact than the reference design components;
    - It conforms to major international codes.

- Addition of alternative systems (A-2 safety systems)
  The FLSS, FCVS and RDCF are fitted to the UK ABWR as countermeasures against beyond design basis faults and severe accidents as part of the lessons learnt from Fukushima Daiichi Accident. However, their design have been adapted so that they could also serve as backup systems in the combined event of a reactor transient (frequent fault excluding accidents such as LOCAs) and the failure of the Safety Category and Class A-1 ESFs (ECCS and RHR for containment heat removal) to perform their safety functions for design basis events, such as a Station Blackout (SBO) – LOOP and failure of the Safety Class 1 EDGs. Also, the AC, which main purpose is to control primary containment atmosphere composition and pressure during plant power operation, has been adapted to be able to perform containment venting for long term heat removal, thus providing “N+1 redundancy” for this function with the FCVS. Therefore the following design considerations have been implemented:

    - Diverse design relatively to their equivalent Safety Category and Class A-1 safety systems in terms of structure and components, operating conditions and functioning principles; and physical separation between A-1 and A-2 to ensure the A-2 safety systems will not fail for the same cause as the A-1 safety systems.
    - Implementation of logic for automatically actuating the FLSS and the RDCF on fault occurrence and detection of failure of the A-1 safety systems. For the FCVS and the AC, for which a short response is not required, operation is done manually from the control panels.
    - Provision of A-2 control panels in the Control Building (C/B) so that operation of these systems can be performed either in the MCR or the B/B Control Room.
    - Addition of a hardened vent line for the AC to the main stack, and implementation of a control logic to permit opening of the venting valves event if the containment isolation signal was transmitted.

The ECCS and containment cooling systems of the UK ABWR are active systems. A justification that these active systems satisfy the ALARP principle, when compared with the potential use of alternative passive system options that are used on some other reactor designs, is presented in [Ref-13.6-4]. As described above, the FCS has been modified by introducing PARs to enhance hydrogen control in the PCV, based on an ALARP assessment in [Ref-13.6-5]. The same reference has also resulted in the addition of PARs in the Reactor Building to satisfy the ALARP principle.

An ALARP discussion of potential design options to minimise primary containment leakage is presented in [Ref-13.6-6]. This has resulted in a change to the material used for the drywell head flange gasket for the UK ABWR in comparison to the Japanese reference design.

[Ref-13.6-7] presents a discussion which compares methods of capturing or reducing fission products inside the primary containment with alternative methods used in other reactor types to determine relevant good practice. This concludes that the UK ABWR design satisfies the ALARP principle without changes from the Japanese reference design.

[Ref-13.6-8] presents an ALARP discussion which compares methods of providing water cooling for the RCCV head with other reactor types to determine relevant good practice. This has resulted in a number of design changes in comparison to the Japanese reference design. An example is to increase the extent of Post Weld Heat Treatment of the drywell head region.

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-13.6-12]. 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.

Thus, application of the general ME process described in [Ref-13.6-11] has demonstrated that the UK ABWR mechanical design of the SSCs within the scope of this chapter achieves risk levels that are tolerable and ALARP, when taking account of the relevant GDA design changes.

There are many interfaces between ECCS components and the reactor coolant water. The chemistry regime for the reactor coolant system is optimised to provide the best balance between reducing risks of degradation of interfacing SSCs and minimising worker dose rates to levels that are ALARP. Further details are given in Chapter 23 : Reactor Chemistry.

13.7 Conclusions

This chapter provides a high-level demonstration that the ESFs for the UK ABWR have been designed to the highest standards consistent with their safety critical role for the UK ABWR. ESFs provide an independent means from the normally operating systems (described in Chapter 17) for providing the safety functions of (1) fuel cooling, (2) long term heat removal and (3) confinement of radioactive materials. ESF consists of core cooling systems (ECCS and the alternative systems for core cooling), and Primary and Secondary Containment Facilities. The approach described in this chapter is to show that the ESF systems have been designed to provide a safe response to the most bounding accidents while also providing a safe response to the wider range of higher frequency but lower consequence accidents. ESF design for the UK ABWR is based on modern codes and standards (Chapter 5.8) and evolutionary developments from more than 50 years of experience of designing, constructing, commissioning and supporting the operation of BWRs and ABWRs.

As described in this chapter the core cooling systems consist of the following:

- ECCS (RCIC/HPCF, ADS, LPFL)
- Alternative systems for core cooling (FLSS. RDCF)

These ECCS subsystems are designed to provide high pressure make-up water (RCIC/HPCF) at full reactor transient pressures and to manage, using the ADS, the safe transition to the lower pressure safety functions of the LPFL/RHR. All three subsystems can perform their safety functions even in the presence of a major fault and equipment being unavailable due to planned maintenance. On the other hand, the FLSS and the RDCF are designed to provide reactor depressurisation and low-pressure injection in the case the ECCS was unavailable due to common cause failure of the ECCS divisions in spite of the redundancy provided. However, despite the considerable fault tolerance of each core cooling subsystem, the use of high quality and high availability components are an integral part of the design to ensure the maximum availability of equipment if the core cooling systems are required to respond to an accident condition.

The Primary and Secondary Containment Facilities consist of the following:

- Primary Containment Vessel (PCV)
- Primary Containment Isolation System (PCIS)
- Primary Containment Vessel Gas Control System (FCS, AC)
- Containment Heat Removal System (RHR, AC, FCVS)
- Reactor Building (R/B)
- Standby Gas Treatment System (SGTS)

The structural integrity aspects of the design of the PCV and R/B are provided in Chapters 8 and 10 respectively. This chapter has demonstrated that the systems listed above act in an integrated way with the ECCS to ensure that there are multiple independent barriers to prevent the release of radioactivity. This chapter and its supporting references show that the risk of major failure of one or more of the ESF subsystems is very low and the design of the ESF is fully aligned with UK good practice. The ESF described in this chapter, supported by Chapters 8, 10, 24, 25 and 26, show that the risks of major failure are as low as is reasonably practicable.

13.8 References

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[Ref-13.3-18] Hitachi-GE Nuclear Energy, Ltd., " Basis of Safety Cases on Reinforced Concrete Containment Vessel (RCCV)", GA91-9201-0002-00007 (DD-GD-0003) Rev.7, June 2017
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[Ref-13.4-2] Hitachi-GE Nuclear Energy, Ltd., "Reactor Core Isolation Cooling System System Design Description", GE51-1001-0001-00001 (SD-GD-0008) Rev.2, June 2017
[Ref-13.4-3] Hitachi-GE Nuclear Energy, Ltd., "Reactor Core Isolation Cooling System P&ID (1/2)", GE51-2101-0001-00001 (310QC98-319) Rev.2, June 2017
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[Ref-13.4-9] Hitachi-GE Nuclear Energy, Ltd., "Residual Heat Removal System P&ID (1/3)", GE11-2101-0001-00001 (310QC98-312) Rev.2, June 2017
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[Ref-13.6-5] Hitachi-GE Nuclear Energy, Ltd., “ALARP discussion on flammable gas control”, GA91-9201-0003-00740 (AE-GD-0438), Rev 2, April 2017
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[Ref-13.6-7] Hitachi-GE Nuclear Energy, Ltd., “ALARP discussion on methods of further capturing or reducing fission products inside containment”, GA91-9201-0003-00914 (AE-GD-0447), Rev 0, October 2015
[Ref-13.6-8] Hitachi-GE Nuclear Energy, Ltd., “ALARP discussion of provision of water cooling for the RCCV drywell head”, GA91-9201-0003-01169 (DE-GD-0056), Rev 0, February 2016
[Ref-13.6-9] Hitachi-GE Nuclear Energy, Ltd., “Basis of Safety Cases on Emergency Core Cooling systems”, GA91-9201-0002-00020 (SE-GD-0164), Rev 2, June 2017
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Appendix A: Safety Functional Claims Table

(1) SFC Table of PCIS and PCV

	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
1	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 the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.	A	1
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 line 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	NB SFC 4-7.3	The NB 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
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 - 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	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	4-7	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 line 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	RHR SFC 4-7.1	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.	A	1
5	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault (break of RCIC steam supply pipe outside the PCV only)	Fault Conditions	RCIC SFC 4-7.1	The RCIC 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

	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		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	HPCF SFC 4-7.1	The HPCF 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
7	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
8	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
9	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 7.2 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3	LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line 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	AC SFC 4-7.1	The AC 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
10	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	FCVS SFC 4-7.1	The FCVS 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
11	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 line 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	RCW SFC 4-7.1	The RCW 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

	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
12	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	IA SFC 4-7.1	The IA 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
13	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 line 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	SA SFC 4-7.1	The SA 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
14	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	HPIN SFC 4-7.1	The HPIN 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
15	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 line 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	RD SFC 4-7.1	The RD 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
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 - FDW line 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	SAM SFC 4-7.1	The SAM 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
17	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	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line LOCA - MS line break LOCA - RHR outlet line break	Fault Conditions	SPCU SFC 4-7.1	The SPCU 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

	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					10.1 10.2 10.3	LOCA outside PCV - MS line break LOCA outside PCV - CUW line break LOCA outside PCV - FDW line (RCIC connected) break					
18	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	RDCF SFC 4-7.1	The RDCF 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
19	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 line 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	PRM SFC 4-7.1	The PRM mechanical 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
20	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 line 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	HNCW SFC 4-7.1	The HNCW 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
21	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	MUWC SFC 4-7.1	The MUWC 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
22	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	FLSS SFC 4-7.1	The FLSS 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
23	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	FLSR SFC 4-7.1	The FLSR 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

	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
24	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	<u>Fault Conditions</u>	RVI SFC 4-7.1	The RVI mechanical 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
25	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	<u>Fault Conditions</u>	PCV L/T SFC 4-7.1	The PCV leak test facility 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
26	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	<u>Fault Conditions</u>	CAMS SFC 4-7.1	The CAMS mechanical 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
27	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 line 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	<u>Fault Conditions</u>	TIP SFC 4-7.1	The TIP mechanical 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
28	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	<u>Fault Conditions</u>	MUWP SFC 4-7.1	The MUWP 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
29	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 line 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	<u>Fault Conditions</u>	VGL SFC 4-7.1	The VGL 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

	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
30	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 line 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	ANI SFC 4-7.1	The ANI 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
31	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 7.2 8.1 8.2 9.1.1 9.1.2 9.2 9.3	LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line (LPFL connected) break LOCA - FDW line (RCIC connected) break LOCA - MS line break LOCA - RHR outlet line break	Fault Conditions	PCV SFC 4-7.1	Any steam released into the RCCV from a possible pipe rupture in the primary system will be condensed by the S/P, and any significant pressure rise will be suppressed	A	1
32	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	7.1 7.2 8.1 8.2 9.1.1 9.1.2 9.2 9.3	LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line (LPFL connected) break LOCA - FDW line (RCIC connected) break LOCA - MS line break LOCA - RHR outlet line break	Fault Conditions	PCV SFC 4-7.2	The RCCV can withstand the maximum excessive pressure and temperature caused by the defined LOCA events including piping break such as instantaneous, complete and double-ended guillotine break of one feedwater piping or one main steam piping	A	1
33	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	2.2 2.3 3.1 4.6 5.1 7.1 7.2 8.1 8.2 9.1.1 9.1.2 9.2 9.3 2.2.1 2.3.1 5.1.1 5.1.2 5.1.3 3.1.1	Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Radiation monitor failure Short term LOOP LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line (LPFL connected) break LOCA - FDW line (RCIC connected) break LOCA - MS line break LOCA - RHR outlet line break Reactor pressure regulator failure in the open direction Loss of main condenser vacuum with failure of Class 1 Reactivity Control function Short-term LOOP with CCF of EDGs Short-term LOOP with CCF of initiation signal Short-term LOOP with digital CCF Loss of all feedwater flow with failure of Class 1 fuel	Normal and Fault Conditions	PCV SFC 4-7.3	The air leakage ratio of the RCCV is based on 0.4 percent per day or less of free volume of the containment at ordinary temperature and with a 90 percent of the maximum design pressure	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					 5.2.1 5.2.2 5.2.3 11.2 11.11 11.12 18.1	cooling function Medium term LOOP with CCF of EDGs Medium term LOOP with CCF of initiation signal Medium term LOOP with digital CCF Inadvertent opening of all ADS Loss of all RSW Loss of Class 1 HVAC Loss of Ultimate Heat Sink					
34	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	2.2 2.3 3.1 4.6 5.1 7.1 7.2 8.1 8.2 9.1.1 9.1.2 9.2 9.3 10.1 10.2 10.3 2.2.1 2.3.1 5.1.1 5.1.2 5.1.3 3.1.1 5.2.1 5.2.2 5.2.3 11.2 11.11 11.12 18.1	Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Radiation monitor failure Short term LOOP LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line (LPFL connected) break LOCA - FDW line (RCIC connected) break 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 Reactor pressure regulator failure in the open direction Loss of main condenser vacuum with failure of Class 1 Reactivity Control function Short-term LOOP with CCF of EDGs Short-term LOOP with CCF of initiation signal Short-term LOOP with digital CCF Loss of all feedwater flow with failure of Class 1 fuel cooling function Medium term LOOP with CCF of EDGs Medium term LOOP with CCF of initiation signal Medium term LOOP with digital CCF Inadvertent opening of all ADS Loss of all RSW Loss of Class 1 HVAC Loss of Ultimate Heat Sink	Normal and Fault Conditions	PCV SFC 4-7.4	The RCCV and structures within the RCCV have a structural strength that maintains integrity when assumed static load and dynamic load generated in normal / fault conditions are appropriately combined with the relevant seismic load	A	1
35	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	2.2 2.3 3.1 4.6 5.1 7.1 7.2 8.1	Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Radiation monitor failure Short term LOOP LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break	Normal and Fault Conditions	PCV SFC 4-7.5	As for the steel parts in the RCCV, brittle fracture is prevented by taking the lowest design temperature (10°C) into consideration	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					8.2	LOCA - LPFL line break					
					9.1.1	LOCA - FDW line (LPFL connected) break					
					9.1.2	LOCA - FDW line (RCIC connected) break					
					9.2	LOCA - MS line break					
					9.3	LOCA - RHR outlet line break					
					10.1	LOCA outside PCV - MS line break					
					10.2	LOCA outside PCV - CUW line break					
					10.3	LOCA outside PCV - FDW line (RCIC connected) break					
					2.2.1	Reactor pressure regulator failure in the open direction					
					2.3.1	Loss of main condenser vacuum with failure of Class 1 Reactivity Control function					
					5.1.1	Short-term LOOP with CCF of EDGs					
					5.1.2	Short-term LOOP with CCF of initiation signal					
					5.1.3	Short-term LOOP with digital CCF					
					3.1.1	Loss of all feedwater flow with failure of Class 1 fuel cooling function					
					5.2.1	Medium term LOOP with CCF of EDGs					
					5.2.2	Medium term LOOP with CCF of initiation signal					
					5.2.3	Medium term LOOP with digital CCF					
					11.2	Inadvertent opening of all ADS					
					11.11	Loss of all RSW					
					11.12	Loss of Class 1 HVAC					
					18.1	Loss of Ultimate Heat Sink					
36	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	2.2	Reactor pressure regulator failure in the open direction	<u>Fault Conditions</u>	PCV SFC 4-7.6	The RCCV and structures within the RCCV have sufficient structural strength to maintain integrity against the following hydrodynamic loads. - Gas / steam release - Pool swell - Steam condensation (oscillation / chugging loads) - Annulus Pressurisation	A	1
					2.3	Loss of main condenser vacuum					
					3.1	Loss of all feedwater flow					
					4.6	Radiation monitor failure					
					5.1	Short term LOOP					
					7.1	LOCA - RPV bottom drain line break					
					7.2	Small line break LOCA					
					8.1	LOCA - HPCF line break					
					8.2	LOCA - LPFL line break					
					9.1.1	LOCA - FDW line (LPFL connected) break					
					9.1.2	LOCA - FDW line (RCIC connected) break					
					9.2	LOCA - MS line break					
					9.3	LOCA - RHR outlet line break					
					10.1	LOCA outside PCV - MS line break					
					10.2	LOCA outside PCV - CUW line break					
					10.3	LOCA outside PCV - FDW line (RCIC connected) break					
					2.2.1	Reactor pressure regulator failure in the open direction					
					2.3.1	Loss of main condenser vacuum with failure of Class 1 Reactivity Control function					
					5.1.1	Short-term LOOP with CCF of EDGs					
					5.1.2	Short-term LOOP with CCF of initiation signal					
					5.1.3	Short-term LOOP with digital CCF					
					3.1.1	Loss of all feedwater flow with failure of Class 1 fuel cooling function					

	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.5.6 Table 5.6-1: High level safety functions in UK ABWR		Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					5.2.1 5.2.2 5.2.3 11.2 11.11 11.12 18.1	Medium term LOOP with CCF of EDGs Medium term LOOP with CCF of initiation signal Medium term LOOP with digital CCF Inadvertent opening of all ADS Loss of all RSW Loss of Class 1 HVAC Loss of Ultimate Heat Sink					
37	5	Others	5-17	Function to provide structural support to SSCs	2.2 2.3 3.1 4.6 5.1 7.1 7.2 8.1 8.2 9.1.1 9.1.2 9.2 9.3 10.1 10.2 10.3 2.2.1 2.3.1 5.1.1 5.1.2 5.1.3 3.1.1 5.2.1 5.2.2 5.2.3 11.2 11.11 11.12 18.1	Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Radiation monitor failure Short term LOOP LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line (LPFL connected) break LOCA - FDW line (RCIC connected) break 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 Reactor pressure regulator failure in the open direction Loss of main condenser vacuum with failure of Class 1 Reactivity Control function Short-term LOOP with CCF of EDGs Short-term LOOP with CCF of initiation signal Short-term LOOP with digital CCF Loss of all feedwater flow with failure of Class 1 fuel cooling function Medium term LOOP with CCF of EDGs Medium term LOOP with CCF of initiation signal Medium term LOOP with digital CCF Inadvertent opening of all ADS Loss of all RSW Loss of Class 1 HVAC Loss of Ultimate Heat Sink	<u>Normal and Fault Conditions</u>	PCV SFC 5-17.1	The RCCV has a structural strength that maintains integrity when assumed static load and dynamic load generated in fault conditions (and in normal condition) are appropriately combined with the relevant seismic load to support SSCs within RCCV (and/or to support the RCCV)	A	1
38		Others	5-17	Function to provide structural support to SSCs	7.1 7.2 8.1 8.2 9.1.1 9.1.2 9.2 9.3	LOCA - RPV bottom drain line break Small line break LOCA LOCA - HPCF line break LOCA - LPFL line break LOCA - FDW line (LPFL connected) break LOCA - FDW line (RCIC connected) break LOCA - MS line break LOCA - RHR outlet line break	<u>Fault Conditions</u>	PCV SFC 5-17.2	The RCCV and structures within the RCCV have sufficient structural strength to maintain integrity against the following hydrodynamic loads to support SSCs (and/or to support the RCCV). - Gas / steam release - Pool swell - Steam condensation (oscillation / chugging loads) - Annulus Pressurisation	A	1

(2) SFC Table of PCV Gas Control Systems

	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
1	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 9.1.1 9.1.2 9.2 9.3	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line (LPFL connected) break LOCA - FWD line (RCIC connected) break LOCA - MS line break LOCA - RHR line break	Fault Conditions	[FCS SFC 5-15.1]	The FCS backs up the confinement function by maintaining the hydrogen and oxygen concentrations in the PCV below the flammability limits through recombination of the gases generated and accumulated in the PCV that might occur after design basis faults such as LOCA.	B	2
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 line 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	[AC SFC 4-7.1]	The AC 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
3	4	Confinement/ Containment of radioactive materials	4-17	Functions to maintain PCV atmosphere in an inert state for preventing hydrogen combustion	-	No corresponding fault	Normal Conditions	[AC SFC 4-17.1]	The AC supplies nitrogen gas to maintain inert condition within the PCV during inerting operation and plant power operation, which prevents hydrogen combustion in case of design basis faults.	C	3
4	4	Confinement/ Containment of radioactive materials	4-17	Functions to maintain PCV atmosphere in an inert state for preventing hydrogen combustion	-	No corresponding fault	Normal Conditions	[AC SFC 4-17.2]	The AC supplies nitrogen gas to maintain slight positive pressure within the PCV during plat power operation, which prevents air in-leak from secondary containment to the PCV.	C	3
5	5	Others	5-13	Auxiliary functions for plant operation	-	No corresponding fault	Normal Conditions	[AC SFC 5-13.1]	The AC supports the pneumatic equipment inside the PCV requiring nitrogen gas supply such as the SRV accumulators, the inboard MSIV accumulators, instrumentation, etc., and equipment using nitrogen gas inside the R/B to deliver their respective safety functions by supplying nitrogen gas through the HPIN.	C	3
6	3	Long term heat removal	3-2	Function of alternative containment cooling and decay heat removal	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	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 power Inadvertent reactor SCRAM (CRD pump trip)	Fault Conditions	[AC SFC 3-2.1]	The AC is a secondary means to deliver long term PCV heat removal and overpressure protection in the event of design basis faults where the primary long-term containment heat removal means (RHR) has failed.	A	2

	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					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					
					10.4	Small line break LOCA outside PCV					
					11.8	M/C power supply failure on electrical CCF					
					11.9	D/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					
					13.6	Long term SBO					
					13.7	Draindown due to valve failure in RHR					
					13.12	RPV draindown by CUW					
					18.1	Loss of ultimate heat sink					
					18.2	10-3/y earthquake					
					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					

(3) SFC Table of Containment Heat Removal 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-13.1-1]		State	Claim ID	SFC Contents	Cat.	Class
1	3	Long term heat removal	3-1	Functions to remove residual heat after shutdown	1.1	Generator load rejection	<u>Fault Conditions</u>	[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 and infrequent faults such as Anticipated Transient Without Scram (ATWS).	A	1
					1.2	Partial loss of reactor flow (trip of 4 RIPs)					
					1.3	Loss of reactor flow (trip of all RIPs)					
					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					
					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					

	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-13.1-1]		State	Claim ID	SFC Contents	Cat.	Class
2	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 2.1 2.2 2.3 2.4 3.1 4.2 4.4 4.5 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 13.3 13.5 13.7 13.8 13.9 13.10 13.11 13.12 13.13 13.14 13.15 13.16 17.1 17.2	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 Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Radiation monitor failure Loss of all feedwater flow Control rod withdrawal error at power Inadvertent reactor SCRAM (CRD pump trip) SRNM or APRM sensor 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 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 LOCA outside PCV - small line break Inadvertent opening of all ADS Inadvertent start up of all injection system SD- Loss of operating RHR and the same ECCS div. SD - LOOP SD - Draindown due to operating RHR valve failure SD - LOCA inside PCV – FWD line SD - LOCA inside PCV – RHR suction line SD - LOCA inside PCV – LPFL injection line SD - LOCA below TAF SD - RPV draindown by CUW SD - Leakage during FMCRD inspection SD - Leakage during IMC nozzle replacement SD - Leakage during RIP inspection SD - Refuelling bellow perforation Internal fire in R/B Internal fire in Hx/B	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 such as main condenser unavailability and infrequent faults such as LOCA.	A	1

	Top Claim for Mechanical System: Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)		Safety Functional Claim for the Mechanical System and Components (SFC)				
	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-13.1-1]		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					
3	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	<u>Fault Conditions</u>	[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
4	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault.	<u>Fault Conditions</u>	[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
5	3	Long term heat removal	3-2	Function of alternative containment cooling and decay heat removal	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 5.1 5.2 5.3 6.1 7.2 10.4 11.8 11.9 11.10 11.11 11.12 13.6 13.7 13.12	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 power Inadvertent reactor SCRAM (CRD pump trip) Short LOOP Medium LOOP Long LOOP Inadvertent opening of a SRV LOCA - small line break Small line break LOCA outside PCV 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 Long term SBO Draindown due to valve failure in RHR RPV draindown by CUW	<u>Fault Conditions</u>	[AC SFC 3-2.1]	The AC is a secondary means to deliver long term PCV heat removal and overpressure protection in the event of design basis faults where the primary long-term containment heat removal means (RHR) has failed.	A	2

	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-13.1-1]		State	Claim ID	SFC Contents	Cat.	Class
					18.1	Loss of ultimate heat sink					
					18.2	10-3/y earthquake					
					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					
6	3	Long term heat removal	3-2	Function of alternative containment cooling and decay heat removal	1.1	Generator load rejection	Fault Conditions	[FCVS SFC 3-2.1]	The FCVS is a secondary means to deliver long term PCV heat removal and overpressure protection in the event of design basis faults where the primary long-term containment heat removal means (RHR) has failed.	A	2
					1.2	Partial loss of reactor flow (trip of 3 RIPs)					
					1.3	Loss of reactor flow (trip of all RIPs)					
					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					
					3.1	Loss of all feedwater flow					
					4.2	Control rod withdrawal error at power					
					4.4	Inadvertent reactor SCRAM (CRD pump trip)					
					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					
					10.4	Small line break LOCA outside PCV					
					11.8	M/C power supply failure on electrical CCF					
					11.9	D/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					
					13.6	Long term SBO					
					13.7	Draindown due to valve failure in RHR					
					13.12	RPV draindown by CUW					
					18.1	Loss of ultimate heat sink					
					18.2	10-3/y earthquake					
					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					

(4) SFC Table of DWC

	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
1	5	Others	5-18	Function to maintain internal building environment appropriate for SSCs	-	No corresponding fault	Normal Conditions	[DWC SFC 5-18.1]	The DWC controls the design environmental parameters inside the served areas.	C	3

(5) SFC Table of SGTS

	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
1	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Normal Conditions	[SGTS SFC 4-7.1]	The SGTS constitutes a part of the confinement function by filtering the gaseous effluent from the Primary Containment or from the Secondary Containment when required to limit the discharge of radioactive material to the environment.	C	3
2	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 7.1 7.2 8.1 8.2 9.1 9.2 9.3 10.4 11.3 11.10 11.11 11.12 13.5 13.5 13.5 13.5 13.16 14.5 14.7 14.8 17.6 18.1 18.2 18.3	(Not claimed for the design basis) Inadvertent MSIV closure Reactor pressure regulator failure in the open direction Loss of main condenser vacuum Loss of all feedwater flow Radiation monitor failure Short term Loss of off-site power LOCA - RPV bottom drain line break LOCA - small line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line LOCA - MS line break LOCA -RHR Outlet line break Small line break LOCA outside the PCV Inadvertent start-up all injection system Loss of all RCW Loss of all RSW Loss of Class 1 HVAC SD- Short LOOP SD - Medium LOOP SD - Long LOOP Long term LOOP Refuelling bellows perforation Fuel Drop Drop of heavy equipment into the Core Drop of heavy equipment into the SFP Turbine missile Loss of Ultimate Heat Sink 10-3/year Earthquake Design Basis Earthquake (DBE)	Fault Conditions	[SGTS SFC 4-7.2]	The SGTS constitutes a part of the confinement function in the event of design basis faults such as LOCA, radioactive releases from refuelling operations, etc. by maintaining a negative pressure in the Secondary Containment relative to the outdoor atmosphere, and by filtering radiological effluents from the Primary Containment that leak into the Secondary Containment to control and reduce the release of radioactive substances to the environment.	B	2

(6) **SFC Table of ECCS**

	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
1	2	Fuel Cooling	2-1	Functions to cool reactor core	1.1	Generator load rejection	Fault Conditions	[RCIC SFC 2-1.1]	The RCIC is a principal means to provide reactor core cooling as part of the ECCS when the RPV is in high pressure state and in the interval it is being depressurised 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 frequent faults such as loss of the normal feedwater supply and infrequent faults such as LOCA.	A	1
					1.2	Partial loss of reactor flow (trip of 3 RIPs)					
					1.3	Loss of reactor flow (trip of all RIPs)					
					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					
					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					
					4.6	Radiation monitor 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 (RCIC connected) break					
					10.4	LOCA outside PCV – small line break					
					11.2	Inadvertent opening of all ADS					
					11.3	Inadvertent start up of all injection system					
					11.8	M/C power supply failure (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 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					

	Top Claim for Mechanical System: Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)		Safety Functional Claim for the mechanical System and Components (SFC)				
	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					18.3	DB earthquake					
2	2	Fuel Cooling	2-1	Functions to cool reactor core	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 8.1 8.2 9.1 9.2 9.3 10.1 10.2 10.3 10.4 11.3 17.1 17.2 17.3 17.4 17.5 17.6 18.1 18.2 18.3	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 power 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 - small 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 (RCIC connected) break LOCA outside PCV – small line break Inadvertent start up of all injection system 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	Fault Conditions	[HPCF SFC 2-1.1]	The HPCF is a principal means to provide reactor core cooling as part of the ECCS when the RPV is in high pressure state and in the interval it is being depressurised 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 frequent faults such as loss of the normal feedwater supply and infrequent faults such as LOCA.	A	1
3	2	Fuel Cooling	2-1	Functions to cool reactor core	7.1 8.1 8.2 9.1.1	LOCA - RPV bottom drain line break LOCA - HPCF line break LOCA - LPFL line break LOCA - FWD line (LPFL connected) break	Fault Conditions	[HPCF SFC 2-1.2]	The HPCF 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	A	1

	Top Claim for Mechanical System: Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)		Safety Functional Claim for the mechanical System and Components (SFC)				
	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					9.1.2 9.2 9.3 10.1 10.2 10.3 11.2 13.3 13.5 13.7 13.8 13.9 13.10 13.11 13.12 13.13 13.14 13.15 13.16 17.1 17.2 17.3 17.4 17.5 17.6 18.1 18.2 18.3	LOCA - FWD line (RCIC 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 SD - Loss of operating RHR and the same ECCS div. SD - LOOP SD - Draindown due to operating RHR valve failure 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 SD - RPV draindown by CUW SD - Leakage during FMCRD inspection SD - Leakage during replacement of ICM nozzle SD - Leakage during RIP inspection SD - Refuelling bellow perforation 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			the event of infrequent faults such as LOCA.		
4	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	<u>Fault Conditions</u>	[RHR SFC 2-1.1]	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.	A	1
5	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 break LOCA - MS line break LOCA - RHR line break	<u>Fault Conditions</u>	[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 in the event of LOCA inside 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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
6	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 in the event of LOCA outside the PCV.	A	1
7	2	Fuel Cooling	2-1	Functions to cool reactor core	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	[RCIC SFC 2-1.2]	The RCIC is capable of providing reactor core cooling during at least 8 hours 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 a design basis fault with loss of Class 1 reactor core cooling SSCs due to CCF.	A	1
8	2	Fuel cooling	2-2	Function of alternative fuel cooling	1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 2.1 2.2 2.3 2.4 3.1 3.2 4.2 4.4 5.1 5.2 6.1 7.2 10.4 11.4 11.5 11.8 11.9 11.10 11.11 11.12 13.3 13.4 13.5 13.6 13.7 13.8 13.9	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 Radiation monitor failure Loss of all feedwater flow Inadvertent start up of all injection system Control rod withdrawal error at power 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 Inadvertent opening of all ADS (SSLC failure) Inadvertent MSIV closure (SSLC failure) 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 SD - Loss of operating RHR and same ECCS division SD - Loss of operating RHR (Class 1 controller CCF) SD - LOOP SD - SBO SD - Draindown due to operating RHR valve failure SD - LOCA inside PCV - FDW line SD - LOCA inside PCV - RHR suction line	Fault Conditions	[FLSS SFC 2-2.1]	The FLSS is the secondary means to provide reactor core cooling in order to prevent significant damage to the fuel and minimise the reaction between the fuel cladding and the reactor coolant sufficiently in the event of design basis faults with all the primary reactor core cooling means (ECCS) failed.	A	2

	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					13.10 13.11 13.12 13.13 13.14 13.15 17.1 17.2 17.3 17.4 18.1 18.2	SD - LOCA inside PCV - LPFL return line SD - LOCA (mechanical below TAF) SD - RPV draindown by CUW SD - Leakage during FMCRD inspection SD - Leakage during replacement of ICM nozzle SD - Leakage during RIP inspection 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					
9	2	Fuel cooling	2-2	Function of alternative fuel cooling	1.1 1.2 1.3 1.4 1.5 1.6 1.7 1.8 2.1 2.2 2.3 2.4 3.1 3.2 4.2 4.4 5.1 5.2 6.1 7.2 11.4 11.5 11.8 11.9 11.10 11.11 11.12 17.1 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 Radiation monitor failure Loss of all feedwater flow Inadvertent start up of all injection system Control rod withdrawal error at power Inadvertent reactor SCRAM (CRD pump trip) Short LOOP Medium LOOP Inadvertent opening of a SRV LOCA - small line break Inadvertent opening of all ADS (SSLC failure) Inadvertent MSIV closure (SSLC failure) 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 Internal fire in R/B Loss of Ultimate Heat Sink 10^{-3}/y earthquake	Fault Conditions	[RDCF SFC 2-2.1]	The RDCF is an alternative means to depressurise the RPV in order to provide reactor core cooling in low pressure state with the FLSS in the event of design basis faults where the primary means (ECCS) are not available.	A	2
10	2	Fuel cooling	2-2	Function of alternative fuel cooling	1.4 3.1 5.1 5.2 11.5	Feedwater controller failure - Maximum demand Loss of all feedwater flow Short LOOP with CCF Medium LOOP with CCF Inadvertent MSIV closure (Class 1 SSLC failure)	Fault Conditions	[RDCF SFC 2-2.2]	The RDCF is the principal means to depressurise the RPV in order to provide reactor core cooling in low pressure state with the FLSS after RCIC operation for the first 24hrs in the event of design basis faults such as SBO or Class 1 CCF.	A	2

	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-13.1-1]		State	Claim ID	Claim Contents	Cat.	Class
					11.8 11.9 11.10 11.11 11.12 17.5	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 Internal Missile in the MCR					
11	2	Fuel cooling	2-2	Function of alternative fuel cooling	11.5 11.8 11.9 11.10 11.11 11.12	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	Fault Conditions	[RDCF SFC 2-2.3]	The RDCF with switching valves is the principal means to maintain RPV depressurisation in order to provide reactor core cooling in low pressure state with the FLSS in the event of design basis faults such as SBO or Class 1 CCF after the first 24 hours.	A	3
12	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault (break of RCIC steam supply pipe outside the PCV only)	Fault Conditions	[RCIC SFC 4-7.1]	The RCIC 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
13	4	Confinement/ Containment of radioactive materials	4-7	Functions to confine radioactive materials, shield radiation, and reduce radioactive release	-	No corresponding fault	Fault Conditions	[HPCF SFC 4-7.1]	The HPCF 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
14	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
15	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
16	4	Confinement/ Containment of radioactive materials	4-3	Functions to contain reactor coolant	-	No corresponding fault	Normal Conditions	[RCIC SFC 4-3.1]	The RCIC steam supply line contains reactor coolant (steam) up to the turbine stop valve during power operation. 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
17	4	Confinement/ Containment of radioactive materials	4-4	Functions to contain radioactive material	-	No corresponding fault	Normal Conditions	[RCIC SFC 4-4.1]	The RCIC piping and components outside the RCPB contains radioactive material. Rupture of this piping could lead to a release of radioactive material of dose consequences that are relatively low.	C	3
18	4	Confinement/ Containment of radioactive materials	4-4	Functions to contain radioactive material	-	No corresponding fault	Normal Conditions	[HPCF SFC 4-4.1]	The HPCF piping and components outside the RCPB contains radioactive material. Rupture of this piping could lead to a release of radioactive material of dose consequences that are relatively low.	C	3

Appendix B: Safety Properties Claims Table

The safety properties claims defined for mechanical systems are shown in the following table.

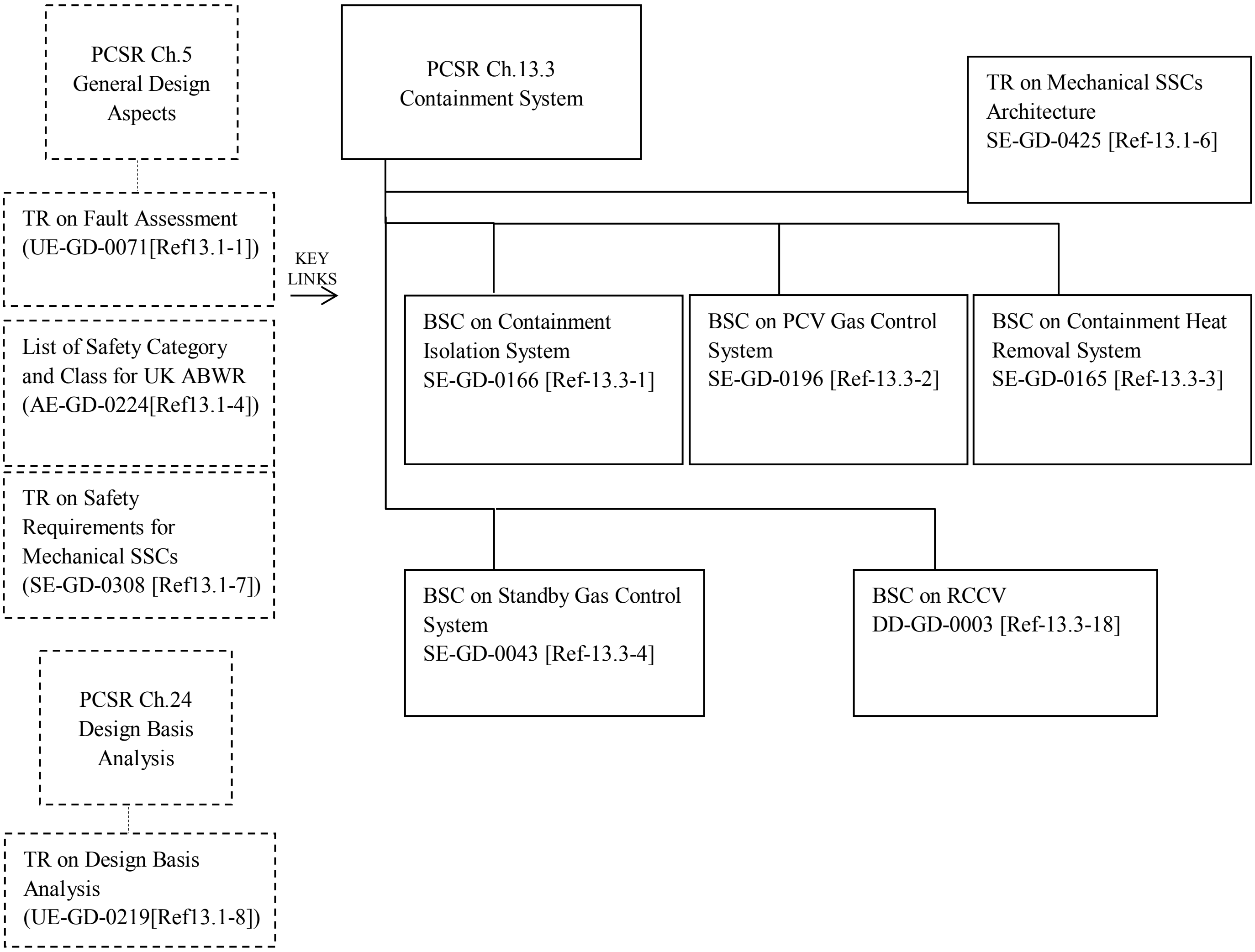
	SPC	Safety Properties Claims (SPC) Contents	SCDM SPC Guide word [Ref-13-1.3]
1	ME SPC1	Design provision against Single Failure Mechanical systems and their support systems are designed with redundancy against single failure of any dynamic component under the worst permissible system availability state 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	Automation 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

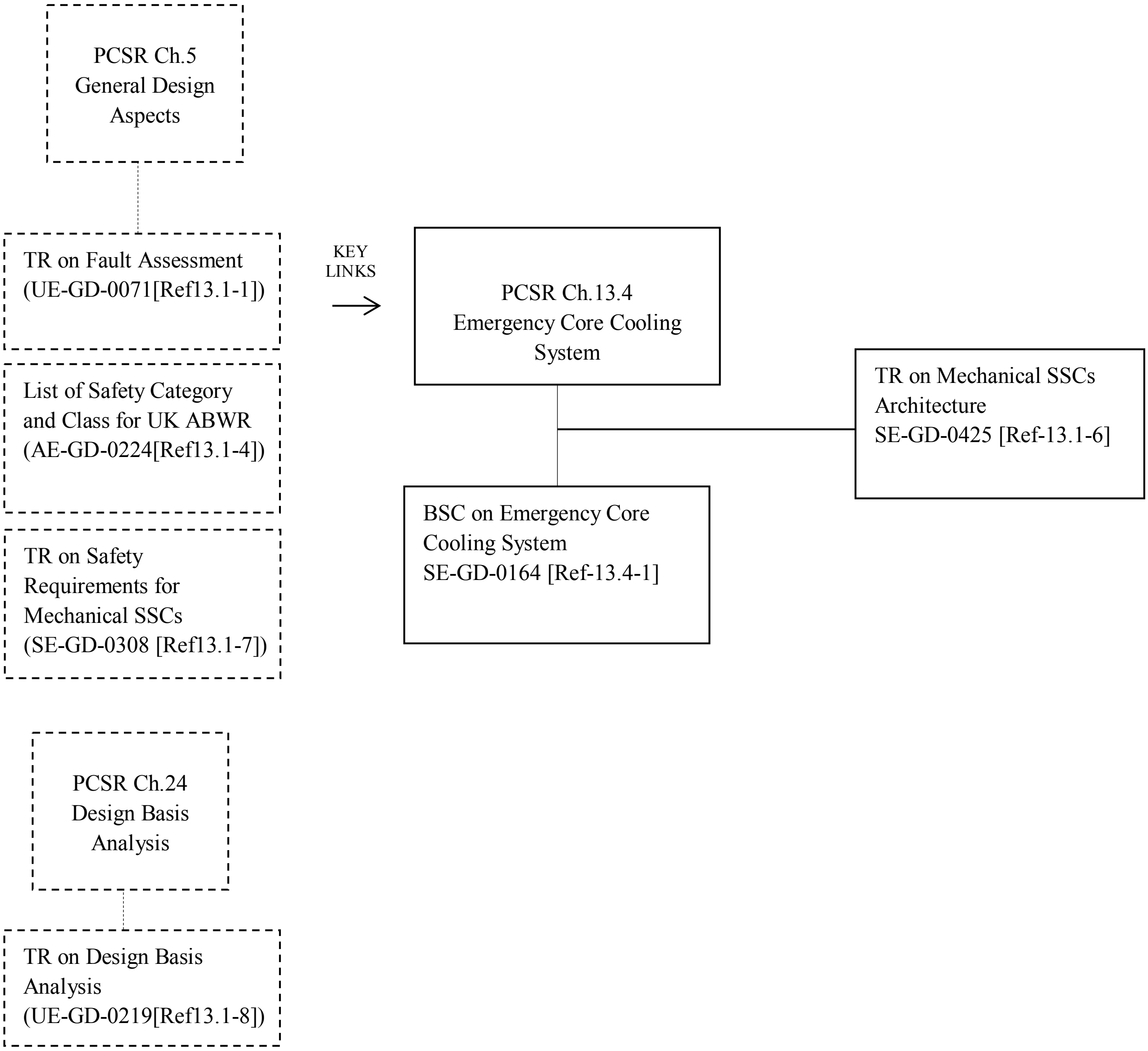
Note: the ME SPCs are derived based on the guide word approach as described in Chapter 5 section 5.3 : General Safety Design Bases.

The safety properties claims table of each system is provided in the following tables.

	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 Automation	ME SPC7 Qualification Provision	ME SPC8 EMIT	ME SPC9 Codes and Standards
PCIS										
Isolation Valves	A-1	X	X	X	X	X	X	X	X	X
PCV Gas Control System										
FCS	B-2	-	-	X	-	X	-	X	X	X
AC Atmosphere Control Portion	C-3	-	-	X	-	X	-	X	X	X
Containment Heat removal System										
RHR for Containment Heat Removal	A-1	X	X	X	X	X	X	X	X	X
Venting System for Containment Heat Removal (AC, FCVS)	A-2	X	X	X	X	X	X	X	X	X
DWC	C-3	-	-	X	-	X	-	X	X	X
DWC	C-3	-	-	X	-	X	-	X	X	X
SGTS										
SGTS	B-2	-	-	X	-	X	-	X	X	X
ECCS										
High Pressure ECCS (RCIC, HPCF)	A-1	X	X	X	X	X	X	X	X	X
Low Pressure ECCS (LPFL)	A-1	X	X	X	X	X	X	X	X	X
ADS	A-1	X	X	X	X	X	X	X	X	X
Alternative System for core cooling (FLSS)	A-2	X	X	X	X	X	X	X	X	X
RDCF	A-2	X	X	X	X	X	X	X	X	X

Appendix C: Document Map




PCSR Ch.5
General Design
Aspects
TR on Fault Assessment
(UE-GD-0071[Ref13.1-1])
List of Safety Category
and Class for UK ABWR
(AE-GD-0224[Ref13.1-4])
TR on Safety
Requirements for
Mechanical SSCs
(SE-GD-0308 [Ref13.1-7])
PCSR Ch.24
Design Basis
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TR on Design Basis
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(UE-GD-0219[Ref13.1-8])
KEY
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PCSR Ch.13.4
Emergency Core Cooling
System
TR on Mechanical SSCs
Architecture
SE-GD-0425 [Ref-13.1-6]
BSC on Emergency Core
Cooling System
SE-GD-0164 [Ref-13.4-1]