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UK HPR1000 GDA Project

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Pre-Construction Safety Report
Chapter 6
Reactor Coolant System

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6.1 List of Abbreviations and Acronyms

ALARP	As Low As Reasonably Practicable
APG	Steam Generator Blowdown System [SGBS]
ARE	Main Feedwater Flow Control System [MFFCS]
ASG	Emergency Feedwater System [EFWS]
BE	Best Estimate
BOL	Beginning Of Life
CAE	Claims, Arguments, Evidence
CB	Core Barrel
CCF	Common Cause Failure
CL	Cold Leg
CRDM	Control Rod Drive Mechanism
CRGA	Control Rod Guide Assembly
CS	Core Support Structure
EMI	Electromagnetic Interference
DBC	Design Basis Condition
DEC	Design Extension Condition
DEC-A	Design Extension Condition A
DEC-B	Design Extension Condition B
DiD	Defence in Depth
DP	Design Pressure
DR	Design Reference
ECS	Extra Cooling System [ECS]
EDG	Emergency Diesel Generator
EHR	Containment Heat Removal System [CHRS]
EMIT	Examination, Maintenance, Inspection and Testing
EOL	End Of Life
FAT	Factory Acceptance Test
F&B	Feed and Bleed
GCT	Turbine Bypass System [TBS]

GDA	Generic Design Assessment
HBSC	Human Based Safety Claims
HFE	Human Factors Engineering
HIC	High Integrity Component
HL	Hot Leg
HMI	Human Machine Interface
HPR1000 (FCG3)	Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3
I&C	Instrumentation and Control
ICIA	In-Core Instrumentation Assembly
IGA	In-Core Instrumentation Guide Assembly
IRWST	In-containment Refuelling Water Storage Tank
IS	Internal Structure
ISI	In-Service Inspection
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MCL	Main Coolant Line
MCR	Main Control Room
MSSV	Main Steam Safety Valve
NDT	Non Destructive Testing
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NSSS	Nuclear Steam Supply System
OPEX	Operating Experience
PCER	Pre-Construction Environmental Report
PCSR	Pre-Construction Safety Report
PRT	Pressuriser Relief Tank
PSA	Probabilistic Safety Assessment
PSI	Pre-Service Inspection
PSR	Preliminary Safety Report
PSV	Pressuriser Safety Valve

PWR	Pressurised Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
PZR	Pressuriser
RBS	Emergency Boration System [EBS]
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant System [RCS]
RCPB	Reactor Coolant Pressure Boundary
RCV	Chemical and Volume Control System [CVCS]
REA	Reactor Boron and Water Makeup System [RBWMS]
REN	Nuclear Sampling System [NSS]
RGL	Rod Position Indication and Rod Control System [RPICS]
RGP	Relevant Good Practice
RHR	Residual Heat Removal
RIC	In-Core Instrumentation System [ICIS]
RIS	Safety Injection System [SIS]
RMI	Reflective Metallic Insulation
RPE	Nuclear Island Vent and Drain System [VDS]
RPV	Reactor Pressure Vessel
RRI	Component Cooling Water System [CCWS]
RT_{NDT}	Reference Nil Ductility Transition Temperature
RVI	Reactor Vessel Internals
SADV	Severe Accident Dedicated Valve
SBO	Station Black Out
SFC	Single Failure Criterion
SG	Steam Generator
SGN	Nitrogen Distribution System [NDS]
SGTR	Steam Generator Tube Rupture
SL	Surge Line
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSS	Standstill Seal System

TEG	Gaseous Waste Treatment System [GWTS]
TH	Thermal Hydraulic
TLOCC	Total Loss of Cooling Chain
UK HPR1000	UK version of the Hua-long Pressurised Reactor
UL	Crossover Leg
USE	Upper Shelf Energy
VDA	Atmospheric Steam Dump System [ASDS]
VVP	Main Steam System [MSS]

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Reactor Coolant System (RCP [RCS]).

6.2 Introduction

The purpose of Pre-Construction Safety Report (PCSR) Chapter 6 is to provide engineering substantiation that the design of Reactor Coolant System (RCP [RCS]) delivers the necessary nuclear safety, in an appropriate manner, depending on the safety function category and safety classification for the UK version of the Hua-long Pressurised Reactor (UK HPR1000). The information presented in this document is based on the version 3 of the UK HPR1000 Design Reference (DR3), as described in UK HPR1000 Design Reference Report (Reference [1]).

6.2.1 Chapter Route Map

The ***Fundamental Objective*** of the UK HPR1000 is presented in Chapter 1 - "*The Generic UK HPR1000 could be constructed, operated, and decommissioned in the UK on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the environment.*"

In order to achieve the fundamental objective, five high level claims and several decoupled low level claims are defined in Chapter 1. The claims related to the Reactor Coolant System (RCP [RCS]) are the Claims of Nuclear Safety, i.e. ***Claim 3*** (Level 1) and ***Claim 3.3*** (Level 2):

a) ***Claim 3****: The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable.*

b) ***Claim 3.3****: The design of the processes and systems has been substantiated and the safety aspects of operation and management have been substantiated.*

Chapter 6 is intended to support **Claim 3.3**. In order to support this level 2 claim, a Level 3 claim is developed for RCP [RCS] as identified below:

Claim 3.3.2: *The design of the Reactor Coolant System has been substantiated.*

Therefore, the main objective of Chapter 6 is to present the information of the UK HPR1000 RCP [RCS] design to support ***Claim 3.3.2***.

According to Reference [2], the trail from safety claims through arguments to evidence shall be clearly set out in the safety case. This chapter is not produced in the form of a strict *Claim-Argument-Evidence* structure. However, a ***Route Map*** intending to set out a "direction of moving forward" for Chapter 6 is identified and presented in Appendix 6A for the future operator.

6.2.2 Chapter Structure

The general structure of this chapter is presented as below:

a) Sub-chapter 6.1 - lists all the abbreviations and acronyms presented in this chapter;

b) Sub-chapter 6.2 - introduces the route map for the RCP [RCS], chapter structure and interfaces with other PCSR chapters;

c) Sub-chapter 6.3 - presents the relative codes and standards which are to be used in the RCP [RCS] and the design of its components;

d) Sub-chapter 6.4 - presents the information for RCP [RCS] system design;

e) Sub-chapter 6.5 - presents the information for RCP [RCS] equipment design;

f) Sub-chapter 6.6 - presents a brief description of overpressure protection and provides links to the detailed analysis reports;

g) Sub-chapter 6.7 - presents the ALARP assessment for the RCP [RCS];

h) Sub-chapter 6.8 - presents a general conclusion for the system and component design;

i) Sub-chapter 6.9 - presents all the documents referenced in this chapter;

j) Appendix 6A - presents the route map for Chapter 6;

k) Appendix 6B - presents the simplified flow diagram of the RCP [RCS];

l) Appendix 6C - presents all the tables;

m) Appendix 6D - presents all the figures.

6.2.3 Interfaces with Other Chapters

The Pre-Construction Safety Report (PCSR) contains various chapters and information on the UK HPR1000 design. A brief process flow chart of the PCSR chapters is presented in Subchapter 1.7 (*Structure and Contents of the PCSR*).

According the process flow chart, Chapter 6 mainly presents the following information:

a) Collects all the safety functional requirements, engineering design principles, and/or other requirements or principles that shall be fulfilled or taken into account during RCP [RCS] design;

b) Provides the design result of the RCP [RCS] and design substantiation (combined with adequate references). The result of RCP [RCS] design presented in this chapter is also used for the safety estimate in various disciplines such as fault analysis, hazard analysis, etc.;

c) Provides supporting functional requirements relevant to safety and operational functions for the interfacing systems, including safety systems, auxiliary systems and secondary loop systems;

d) The ALARP approach presented in Chapter 33 has been applied in Chapter 6 to perform the ALARP demonstration for the structure, system and component designs, which supports the overall ALARP demonstration addressed in Chapter 33.

Moreover, Chapter 6 also provides a brief introduction for the Pre-Construction Environmental Report (PCER) where the system and equipment of the RCP [RCS] need to be discussed.

The relevant interfaces are identified and presented in Appendix 6C Table T-6C-1.

6.3 Applicable Codes and Standards

The general principles relevant to the selection of appropriate standards are presented in PCSR Sub-chapter 4.4.7. Moreover, the principles are presented in detail in References [3] and [4].

Wherever possible, the standards applied for the engineering substantiation shall be:

a) Internationally recognised in the nuclear industry;

b) The latest or currently applicable approved standards; and

c) Consistent with the plant reliability goals necessary for safety, etc.

During GDA, the applicable codes and standards for the design of RCP [RCS] SSCs are identified and used to carry out suitable analysis. Then a compliance analysis is carried out. The main codes and standards used in the design of RCP [RCS] SSCs are presented in the Table T-6C-2 and Reference [5]. The information related to the ALARP demonstration on compliance analysis on RGPs is introduced in sub-chapter 6.7. More detailed information is presented in *ALARP Demonstration for Reactor Coolant System,* Reference [6].

6.4 Description of Reactor Coolant System

The Reactor Pressure Vessel (RPV) is located in the centre of the reactor building. The overall schematic diagram of the RCP [RCS] is presented in Figure F-6D-1 of Appendix 6D. There are three loops of the RCP [RCS] linked to the RPV. The Pressuriser (PZR) is connected to the Hot Leg (HL) of the 3rd loop through the Surge Line (SL). Each loop consists of one Steam Generator (SG), one Reactor Coolant Pump, and Reactor Coolant Pipes, Reference [7].

6.4.1 Safety Requirements

This Sub-chapter identifies all the safety requirements relevant to the RCP [RCS] design, including safety functional requirements and engineering design requirements and principles.

The safety functional requirements are presented in Sub-chapter 6.4.1.1. The design requirements are presented in Sub-chapter 6.4.1.2.

These engineering design requirements and principles are mainly derived from Chapters 4, 15, 18, 19, 30, and 31 of the PCSR. Moreover, there are detailed requirements / principles presented in References [3], [4], [8], [9] and [10].

The safety requirements mentioned above are developed for the UK HPR1000 during the Generic Design Assessment (GDA) and are to be used further during the site licencing stage.

Moreover, an engineering schedule is developed during GDA, Reference [11]. The engineering schedule aims to establish the link between safety assessment and engineering substantiation. SSCs important to safety within GDA scope are selected to validate the methodology of the engineering schedule and provide safety demonstration of ALARP. The requirements of all components of RCP [RCS] will be covered by the engineering schedule at site licensing stage.

6.4.1.1 Safety Functional Requirements

6.4.1.1.1 Control of Reactivity

The design of the RCP [RCS] shall ensure that the reactor coolant water, which is used as a neutron moderator (absorber and reflector), as well as a solvent for enriched boric acid solutions, can provide reactivity control independently from the Rod Cluster Control Assembly (RCCA), Reference [5].

To achieve the reactivity control functions, the RCP [RCS] design ensures to:

a) Contain light water with soluble boron (as required) to serve as the core neutron moderator limiting the velocity of neutrons to the thermal range for reactivity control, Reference [5];

b) Maintain reactivity control within acceptable limits (for example a sub-critical condition in shutdown states) through adding/diluting soluble boron, to compensate for the effects of xenon transients and fuel burn-up, Reference [5];

c) Maintain a uniform concentration of boric acid within the Main Coolant Line (MCL) and PZR, avoiding boron dilution faults in shutdown conditions, Reference [5];

d) Maintain adjustment of the RCP [RCS] boron concentration. The Chemical and Volume Control System (RCV [CVCS]) (supported by the Reactor Boron and Water Makeup System (REA [RBWMS])), Safety Injection System (RIS [SIS]), and Emergency Boration System (RBS [EBS]) contribute to boron concentration adjustment, Reference [5].

6.4.1.1.2 Removal of Heat

The design of the RCP [RCS] shall ensure the heat from the reactor core can be removed to the secondary side systems connected to the SGs, or to the Safety Injection System

(RIS [SIS]) or via Pressuriser Safety Valve (PSV), Reference [5].

To perform the heat removal function, the RCP [RCS] design ensures that:

a) Heat can be transferred to the secondary Steam Generators (SGs):

   1) During plant normal operation, the heat is removed by the Main Steam System (VVP [MSS]), Main Feedwater Flow Control System (ARE [MFFCS]) and Turbine Bypass System (GCT [TBS]), Reference [5];

   2) Under plant accident condition, the heat is removed by the Emergency Feedwater System (ASG [EFWS]) and the Atmospheric Steam Dump System (VDA [ASDS]) if GCT [TBS], ARE [MFFCS] and VVP [MSS] are unavailable, Reference [5].

b) Heat can be removed by the RIS [SIS] or the Containment Heat Removal System (EHR [CHRS]) which is cooled by the Extra Cooling System (ECS [ECS]):

   1) During plant normal startup or shutdown, or during a plant accident when the plant has reached a safe condition, the heat is removed by the RIS [SIS] operating in Residual Heat Removal (RHR) mode, Reference [5];

   2) In the event of total loss of the heat removal features, the RCP [RCS] shall ensure core cooling by primary Feed and Bleed (F&B) operation via Pressuriser Safety Valves (PSVs) combining with RIS [SIS], Reference [5];

   3) Under Total Loss of Cooling Chain (TLOCC) or Station Black Out (SBO) conditions, core residual heat is transferred via the RIS [SIS], EHR [CHRS] combined with the ECS [ECS] to the cool reactor core (see Chapter 7 of the PCSR).

c) Residual heat removal by natural circulation through the core after a loss of primary forced flow, Reference [5];

d) An adequate coolant flow, through the design of the Reactor Coolant Pump, to maintain fuel clad integrity in the event of loss of primary forced flow, Reference [5];

e) Pressure instrumentation and control function via the spray and heater to maintain the pressure of the RCP [RCS] to support the heat removal function, Reference [5].

6.4.1.1.3 Confinement

The design of the RCP [RCS], plus connected system pipework (second barrier), shall ensure confinement of radioactive material, including what results from fuel pin/cladding failure (first barrier) or activation products in the primary coolant (for example, non-condensable gases). The design shall also ensure RCP [RCS] depressurisation to maintain containment integrity (third barrier) under Design Extension Condition B (DEC-B), Reference [5].

To achieve the radioactivity confinement function, the RCP [RCS] design ensures that:

a) PSVs perform the overpressure protection function, thus the maximum pressure and temperature failure limits of the RCP [RCS] will not be exceeded. During plant normal operation, the PSVs are closed and serve as part of the pressure boundary, Reference [5];

b) The Severe Accident Dedicated Valve (SADV) provides defence in depth measures which depressurise the RCP [RCS] to avoid high-pressure melt ejection in DEC-B (severe accident conditions), Reference [5];

c) The pressure retaining boundary of the RCP [RCS] provides reliable integrity and remains intact following a fuel cladding failure (secondary barrier), as well as in DBCs or DECs not induced by a Loss of Coolant Accident (LOCA), Reference [5];

d) The core melt material (corium) retained within the RCP [RCS] is maintained, through the Containment Heat Removal System (EHR [CHRS]), including external RPV cooling (see Chapter 7, Safety systems).

6.4.1.1.4 Extra Safety Functional Requirements

The design of the RCP [RCS] shall ensure the control of reactivity, removal of heat and confinement functions, Reference [5].

To achieve the extra safety functions, the RCP [RCS] design ensures that:

a) Important system operation parameters as well as the status information relevant to the components which perform the safety functions can be monitored and are indicated to the operator;

b) Adequate support for the performance of safety functions is provided, including mechanical support, electric support as well as other support, see sub-chapter 6.5;

c) Preventing, protecting and mitigating hazard impact shall be considered in the RCP [RCS] design, see sub-chapter 6.4.1.2.

6.4.1.2 Design Requirements

These design requirements and principles are mainly derived from Chapters 4, 15, 18, 19, 21, 24, 30, and 31 of the PCSR. Moreover, there are further detailed requirements/principles presented in References [3], [4], [8], [9] and [10].

All of the design requirements are integrated in this Sub-chapter. The preliminary design substantiation is presented in Sub-chapter 6.4.4.

The applicable requirements / principles identified which shall be considered in the RCP [RCS] design are presented below:

a) Safety Classification;

b) Engineering Design Requirements:

   1) The Reliability Design of SSC:

      - Single Failure Criterion (SFC);
      - Independence;
      - Diversity;
      - Fail-Safe;
      - Ageing and Degradation.

   2) Autonomy;

   3) Other design requirements.

c) Equipment Qualification;

d) Protection Design against Internal and External Hazards;

e) Commissioning;

f) Examination, Maintenance, Inspection and Testing (EMIT);

g) Special Thermal-Hydraulic Phenomena;

h) Material Selection;

i) Insulation;

j) Equipment Supplier Design Assurance;

k) Conventional Safety;

l) Human Factors;

m) Radioactive Waste Minimisation;

n) Decommissioning.

6.4.1.2.1 Safety Classification

The aim of the classification is to help ensure that the item is designed, manufactured, constructed, commissioned and operated according to appropriate requirements so as to achieve good quality under all expected operating conditions and realise the safety functions.

As the RCP [RCS] is required to perform safety functions, the safety classification requirements that are summarised in Sub-chapter 4.4 and presented in References [3] and [8] in detail shall be applied to the system and component design.

6.4.1.2.2 Engineering Design Requirements

<u>The Reliability Design of SSC</u>

a) Single Failure Criterion (SFC)

The SFC is used to ensure that more than the minimum numbers of components are provided to carry out a safety function, Reference [3]. The criterion is applicable to a mechanical system which performs a safety function, such that it must be capable of performing its intended safety function in the presence of any single failure. It is beneficial towards ensuring the high reliability of safety systems and to maintaining the plant within its deterministic design basis. The redundancy design helps satisfy this criterion.

The single failure includes active and passive failures:

1) An active single failure is defined as a failure which is sufficient to invalidate the relevant safety function of a component, including the malfunction of a mechanical or electrical component which relies on mechanical movement to complete its intended function upon demand, and the malfunction of an I&C component;

2) A passive single failure is defined as a failure which could occur in a component that does not change its state while realising its function. The passive single failure at the start of a transient should be assessed in an appropriate manner.

The SFC is applied to F-SC1 systems at the system level and F-SC2 systems at the function level, thus redundancy is needed in the design of these systems. Consideration of the single failure criterion at the system level of F-SC1 indicates that these systems must be redundant. Consideration of single failure criteria at the function level for systems fulfilling F-SC2 functions indicates that these systems may not need redundancy.

b) Independence

Independence is accomplished in the design of systems by using functional isolation and/or physical separation, Reference [3].

The following principles for independence should be applied in the design to achieve system reliability and tolerance to faults:

1) Independence between the trains of redundant system should be maintained as far as reasonably practicable (avoidance of common cause failure);

2) Independence between components of different safety categories should be maintained as far as reasonably practicable (avoidance of impact on the component of higher safety category from an item of lower safety category);

3) The components designed to mitigate a potential initiating event should be independent from the effects of this potential initiating event as far as reasonably practicable.

Independence is accomplished in the design of systems by using functional isolation and/or physical separation. Functional isolation is used to reduce adverse effects between elements of connected systems or systems redundantly designed. These adverse effects may be caused by the normal operation, abnormal operation or failure of any part of these systems.

Physical separation should be applied in the layout of systems as far as reasonably practicable, to reduce the potential of common cause failure due to a localised initiating event. The choice of isolation measures (compartmentalisation, distance, orientation etc.) should take into account the nature of the initiating events.

c) Diversity

To reduce the potential for common cause failure, diversity should be realised by incorporating different attributes into the design of systems or components, as appropriate, in the redundant systems or components that perform the same safety function, Reference [3].

In order to achieve the reliability targets and to fulfil the Defence in Depth (DiD) concept, diversity should be realised by incorporating different attributes into the design of systems or components, as appropriate, in the redundant systems or components that perform the same safety function. Such attributes can be different operating principles, different physical variables, different operating conditions, different manufacturers, etc.

Common cause failure of safety measures should be assumed in the analysis for frequent faults. Therefore, a main protection line and a diverse protection line should be established to achieve the fundamental safety objective for frequent faults.

What should be paid special attention is that diversity should be taken into account in the design of systems performing an FC1 or FC2 function based on software to avoid Common Cause Failure (CCF).

d) Fail-Safe

According to Reference [3], the fail-safe requirements shall be considered and incorporated, as appropriate, into the design of systems and components important to safety, so that their failure or the failure of a support feature will not invalidate the performance of the intended safety function.

e) Ageing and Degradation

The general design requirements and management of ageing and degradation are shown in Chapter 4 and Chapter 31 of the PCSR.

Moreover, according to Reference [3], the design life of items important to safety at a Nuclear Power Plant (NPP) shall be determined. Appropriate margins shall be provided in the design to take due account of relevant mechanisms of ageing, neutron embrittlement and wear out and of the potential for age related degradation, to ensure the capability of items important to safety to perform their necessary safety functions throughout their design life. This includes testing, maintenance, maintenance outages, plant states during a postulated initiating event and plant states following a postulated initiating event.

Provision shall be made for monitoring, testing, sampling and inspection to assess ageing mechanisms predicted at the design stage and to help identify unanticipated behaviour of the plant or degradation that might occur in service.

Autonomy

According to Reference [3], the autonomy can be separated into:

a) Autonomy in respect to the operators

If the plant selected parameters exceed set points, the protection system shall come into action, providing automatic scram and initiation of post-trip cooling. The plant shall be designed in such a way that it meets the following autonomy objectives:

1) The numerical targets of DBC-2, DBC-3, DBC-4 and Design Extension Condition A (DEC-A) can be met without operator action from the Main Control Room (MCR) in less than 30 minutes from the first significant signal;

2) The numerical targets of DBC-2, DBC-3, DBC-4 and DEC-A can be met without action outside the MCR in less than 1 hour from the first significant signal;

3) No site based mobile light equipment shall be required in less than 6 hours from accident initiation, for core damage prevention actions in DEC;

4) No site based mobile light equipment shall be required in less than 12 hours from accident initiation, for containment performance assurance in DEC;

5) No offsite or onsite mobile heavy equipment is required in less than 72 hours in both the DBCs and DECs;

6) In addition, the containment system shall be designed in such a way that it can withstand any of the severe accidents considered in DEC, without operator action during the first 12 hours from the beginning of the severe accident conditions.

When extending the timescale in which no operator action is required, the overall safety and practicality of any provisions required should be considered and the emergency response performance that can be expected from operators should be assessed, based on information including the performance achieved in actual major emergencies. When considering extending the autonomy times, this should not be achieved by excessive complication of automatic controls.

Indications of the plant state shall be provided to the operator. It shall be assessed by the designer on a case by case basis whether or not operator overriding of any particular automatic action should be prevented.

The time period from the initiation of any incident condition or accident condition to any serious consequences resulting from the absence of operator intervention (including local actions) shall be as long as practicable.

b) Autonomy in respect to the heat sink

Design provisions shall ensure adequate decay heat removal under DBC and DEC, for 72 hours without external support. The initial means ensuring decay heat removal shall last at least 24 hours.

The design shall include provisions allowing additional means to ensure decay heat removal after 72 hours.

c) Autonomy in respect to power supply systems:

1) Electrical Power Supply

- The period of independence of the installation in relation to external electrical power supplies shall be at least 72 hours; this applies to DBC and DEC;
- The plant shall have an available power supply unit which is independent of the electrical power supply units designed for operational conditions and postulated accidents. It shall have sufficient capacity to support at the same time all these functions: remove decay heat, ensure primary circuit integrity, maintain reactor sub-criticality and monitor the unit state;
- The batteries which perform FC1 and FC2 functions shall be sized so that their expected autonomy is at least 2 hours following any DBC, without recharging;
- In severe accident, the batteries which perform significant safety functions shall meet the requirement that their expected autonomy could be 24 hours without recharging.

2) Compressed Air

Where required to support essential systems, the availability of compressed air reserves should be sufficient to be consistent with the timescale for the availability of the equipment.

Other Design Requirements

a) Prevention of Harmful Interactions of Systems Important to Safety

According to Reference [3], the potential for harmful interactions of systems important to safety at the NPP that might be required to operate simultaneously shall be evaluated and the effects of any harmful interactions shall be prevented.

In the analysis of the potential for harmful interactions of systems important to safety, due account shall be taken of physical interconnections and of the possible effects of one system's operation, mal-operation or malfunction on local environmental conditions of other essential systems, to ensure that changes in environmental conditions do not affect the reliability of systems or components in functioning as intended.

If two fluid systems important to safety are interconnected and are operating at different pressures, either the systems shall both be designed to withstand the higher pressure, or provision shall be made to prevent the design pressure of the system operating at the lower pressure from being exceeded.

b) Considerations Related to the Electrical Power Grid

According to Reference [3], the functionality of items important to safety at the nuclear power plant shall not be compromised by disturbances in the electrical power grid. This requirement shall be considered in the RCP [RCS] design.

6.4.1.2.3 Equipment Qualification

According to Sub-chapter 4.4, equipment qualification is implemented to verify that items important to safety are capable of performing their intended functions when necessary, in the environmental conditions including the variations in ambient environmental conditions that are anticipated in the design for the plant. In order to achieve this objective, the operating conditions considered for equipment qualification include DBCs and DECs.

Equipment qualification includes:

a) Environmental qualification: to verify the performance of the equipment in normal and accidental environmental conditions;

b) Seismic qualification: to verify the performance of the equipment during or after an earthquake.

Considering the results of fault analysis and the safety classifications, the specific equipment to be qualified is listed as follows:

a) Equipment required for environmental qualification:

All normal operational and accident conditions are considered in the equipment qualification process. Normal operational conditions consider the lifetime of the equipment and the environment of the normal condition in the plant where the equipment is placed. The variation in environmental conditions arising from accident conditions is considered in the environmental qualification.

1) Mechanical equipment and electrical equipment that perform FC1 or FC2 functions;

2) Mechanical equipment and electrical equipment that perform FC3 functions required:

   - To maintain a safe state;

   - To protect against DEC-A and mitigate DEC-B.

b) Equipment required for seismic qualification:

The equipment that performs the following functions is seismically qualified: operability (O), functionality (F), integrity (I) or stability (S).

The parameters which are related to the environmental conditions and their impact on equipment are presented below:

c) Temperature

Temperature can indirectly change the performance of the equipment by gradual chemical and physical processes, which is also called thermal aging.

d) Pressure

Pressure and its rapid changes can affect the performance of equipment by exerting additional forces on the equipment. High increase of external or internal pressure may cause structural failure of the fully sealed equipment. The rapid increase of pressure may cause structural failure of the imperfectly sealed equipment.

e) Radiation

Nuclear radiation could induce changes in the atomic and molecular structure of matter through excitation, oxidation, crosslinking, degradation and shearing process, resulting in the change of equipment performance. Some changes improve the performance of the equipment, but most of the changes cause a decline in the performance.

There exist four main types of radiation (α, β, γ and neutron) in nuclear power plants. γ radiation possesses a very high capacity for penetration. On the contrary, the penetration capacity of β radiation is low, 1 mm of steel or 10 mm of water can shield most of the β radiation. The penetration capacity of α radiation is even lower than β radiation. Neutron radiation is considered for equipment near the reactor pit.

f) Humidity

Humidity (high humidity) can directly lead to equipment performance degradation, and can make other environmental conditions worse. For example, moisture could lead to corrosion and current effects at the interfaces of different metals. Moisture could directly reduce the performance of organic materials, degrading their physical, mechanical and electrical performance and deforming them. Moisture on the surface can significantly reduce the insulation resistance and breakdown voltage of the insulation surface.

The environmental conditions of equipment qualification are defined according to the result of fault analysis.

The methods of equipment qualification are presented below:

a) Type test under representative conditions, in accordance with an appropriate test standard;

b) Qualification by analysis:

    1) Calculation (design analysis), usually structural load analysis and mechanical analysis in accordance with an appropriate design code;

    2) Operating Experience (OPEX) based;

    3) Analogy - by comparison with similar qualified equipment.

Considering the specific characteristic of the equipment to be qualified, the methods listed above can be used individually or in combination.

More information related to the equipment qualification method and relevant requirements is presented in Reference [13].

6.4.1.2.4 Protection against Internal and External Hazards

According to Sub-chapter 4.4 and further information presented in Reference [3], the necessary capability, reliability and functionality of items important to safety shall be ensured in the conditions arising from internal and external hazards to deliver relevant safety functions. The design principles relevant to the hazards are presented in Chapters 18 and Chapter 19 of the PCSR. These principles shall be considered in the RCP [RCS] design.

The types of hazards have been identified in Reference [9] for both internal hazards and external hazards. The following types of hazards are applicable for the RCP [RCS]:

a) Applicable types of internal hazards:

    1) Internal Fire;

    2) Internal Flooding;

    3) Internal Explosion;

4) Internal Missile;

5) Dropped Load;

6) High Energy Pipe Failures.

b) Applicable types of external hazards:

1) Earthquakes;

2) Man-made and Industrial Hazards (including aircraft crash and Electromagnetic Interference (EMI)).

6.4.1.2.5 Commissioning

As the RCP [RCS] and its components perform safety functions, these functions shall be effectively demonstrated via commissioning before service.

The commissioning programme phases have been identified for UK HPR1000 in Chapter 30 of the PCSR. The main test stages can be separated as below:

a) Stage I: Preliminary Test Period;

b) Stage II: Functional Tests Period;

c) Stage III: Initial Startup Test Period.

The requirements as well as approach of commissioning presented in Chapter 30 shall be considered in the RCP [RCS] design.

6.4.1.2.6 Examination, Inspection, Maintenance and Testing

According to the requirements which are defined in Sub-chapter 4.4, the design shall be that EMIT activities are facilitated for the purpose of maintaining the capability of SSC important to safety to perform essential safety functions, so as to satisfy the reliability requirement.

The above activities are specified taking into account the design code requirements, reliability analysis and potential degradation mechanisms, commensurate with the safety class of the system. More detailed information is presented in Reference [3].

<u>Examination and Inspection</u>

In-Service Inspection (ISI) is a preventive maintenance process involving the use of Non Destructive Testing (NDT) for pressure mechanical components at scheduled intervals during operation. The ISI is used to detect the anticipated degradation in good time before it compromises structural integrity, and confirm the absence of unanticipated degradation that could lead to failure. More information is presented in Chapter 31 of the PCSR.

<u>Maintenance</u>

According to Chapter 31, maintenance activities are to enhance the reliability of equipment as well. The range of maintenance activities includes servicing, overhaul, repair and replacement of parts, and often, as appropriate, testing, calibration and inspection.

The maintenance types, safety requirements and maintenance strategy are presented in Chapter 31.

Periodic Testing

According to Chapter 31, the periodic test design defines a comprehensive list of the periodic tests that are to be performed on a given system. Each periodic test defines:

a) The test content and scope;

b) The test frequency;

c) The operating mode during which the test is to be performed.

These periodic tests are used to ensure the safety functional availability of a given system. The types of periodic tests, relevant requirements and the methodology of analysing completeness are presented in Chapter 31. These requirements / principles shall be considered in the RCP [RCS] design.

6.4.1.2.7 Special Thermal-Hydraulic Phenomena

Hydraulic phenomena occur during fluid system operation and can be induced by normal or transient operation. Several kinds of hydraulic phenomena may induce potential risk for the safe operation of the facility.

The hydraulic phenomena which shall be considered in the RCP [RCS] design are listed as below:

a) Phenomenon regarding the dead leg;

b) Phenomenon regarding the hot water and cold water mixing;

c) Phenomenon regarding thermal stratification;

d) Phenomenon regarding water hammers;

e) Phenomenon regarding the boiler effect.

6.4.1.2.8 Material Selection

Material selection of systems and equipment is one of the most significant factors for safety and economy to the nuclear power plant. Therefore, special attention shall be paid to material selection at the design stage for SSC to carry out their duties with high reliability throughout the design life of the plant.

The principles and the approach for material selection are presented in Reference [14].

According to Reference [14], the general principles relevant to the material selection of the RCP [RCS] are summarised as below:

a) Material selection shall be consistent with the functional objectives of the system and equipment;

b) Material selection shall be performed in a manner in which the classification shall be reflected; different requirements shall be commensurate with each classification;

c) Materials selected for use in the RCP [RCS] shall be compatible with the full range of environmental conditions which may be encountered over the plant design life;

d) Materials selected for use in the RCP [RCS] shall present high functional reliability and good resistance to aging and degradation throughout the design life to mitigate the risk of performance degradation and failure of SSC;

e) Materials selected for use in the RCP [RCS] shall possess excellent manufacturability, and shall be convenient for performing processing sequences such as forging or casting, machining, heat treatment, welding and inspection;

f) Operating Experience (OPEX) and Feedback shall be taken into account for material selection of the RCP [RCS]. A proven material is preferred, and a novel material (unproven) or a hazardous material is prohibited;

g) Generation and transportation of source terms shall be specially considered when selecting the material to be used in the RCP [RCS]. This is intended to minimise the radiological dose to workers and the public when performing in-service inspection, maintenance, replacement and decommissioning.

Moreover, the water chemistry shall be taken into account when selecting the materials to be used in the equipment design.

6.4.1.2.9 Insulation

During most of the 60-year-long plant life, the RCP [RCS] is kept in operation under high temperature to support the electrical power supply of the plant. Insulation shall be provided for the equipment and the piping system containing or transferring high temperature fluid.

During the equipment and piping system insulation design, the following issues must be considered:

a) During plant normal operation without any maintenance work to be carried out, the insulation design shall reduce the heat loss as much as possible to save energy;

b) During plant maintenance or refuelling, the insulation design shall protect the workers from being scalded;

c) During plant maintenance or refuelling, the insulation design shall ensure the convenience of installation or replacement, especially for the equipment or piping system containing radioactive material;

d) The principles of material selection presented in Sub-chapter 6.4.1.2.8 shall be considered in insulation design. Within the Reactor Building, the material of thermal insulation is RMI. The use of flammable material is prohibited to prevent a potential internal hazard.

6.4.1.2.10 Equipment Supplier Design Assurance

In UK HPR1000 project, the design type of mechanical equipment mainly includes following aspects:

a) CGN carries out the basic and detailed design, and the equipment supplier manufactures the equipment according to the drawings and documents provided by CGN, e.g. Reactor Pressure Vessel;

b) CGN carries out the basic design, publishes the technical specification, qualification requirement and other related documents. The equipment supplier carries out the detailed design and meets the CGN requirements, e.g. pumps, valves, etc.

For the detailed design of equipment completed by a supplier, CGN will supervise the equipment design process and ensure that the design of equipment by the supplier meets the requirements of CGN. The supervisory requirements include:

a) Suitably qualified and experienced person requirement;

b) Prototype design requirement;

c) Prototype qualification requirement;

d) Interface exchange management;

e) Design change management;

f) Supplier documents review management, etc.

The detailed description about design assurance is presented in Reference [15]. Meanwhile, the supplier also needs to consider the impact of the human factors when they carry out the equipment design and manufacture. The related requirements of human factors are presented in equipment specification.

6.4.1.2.11 Conventional Safety

The design of the UK HPR1000 should be developed to eliminate, reduce, isolate or control, so far as is reasonably practicable, the conventional health and safety risks to workers and the public that may arise during the construction, commissioning, operation, maintenance, and decommissioning of the nuclear power plant.

The designers should use the tools of design risk management, such as hazard checklist, hazard Identification workshop and risk assessment steps, in the UK HPR1000 to identify and assess the conventional health and safety risks, as well as eliminate, reduce, isolate and control them by design mitigations. The processes should be recorded in a conventional health and safety design risk register. The conventional health and safety design risk registers for each system and each building in the GDA scope should be developed, and they will be continually developed throughout the lifetime of the design.

The related design processes and requirements of conventional safety are presented in CGN internal management procedure *Construction Design Management Strategy,* Reference [16], and *CDM Design Risk Management Work Instruction*, Reference [17].

6.4.1.2.12 Human Factors

According to Reference [3], a systematic approach needs to be applied to identify the factors that affect human performance and minimise the potential for human error throughout the entire plant lifecycle.

The design needs to allocate functions properly, supports personnel in the fulfilment of their responsibilities and in the performance of tasks. The design also needs to identify human actions that may affect safety and proportionately analyse all tasks important to safety, and limit the likelihood of operational errors and their impact on safety.

A systematic approach on human factors integration is established and applied throughout the entire lifecycle of the UK HPR1000, especially at the design stage. Adequate consideration of human factors is given to ensure that risks from human interactions are managed to a level that is ALARP.

Human factors integration covers the plant locations where operations and maintenance activities take place. To comply with the requirements set above, the following elements will be met:

a) The design should allocate functions properly to minimise the dependence on human actions;

b) Human actions that could impact safety during normal operation, fault and accident conditions should be identified systematically. These human actions important for safety are known as Human Based Safety Claims (HBSC);

c) Appropriate human factors analysis, including task analysis and human reliability analysis, should be performed on the HBSC to identify improvements to systems, procedures or training;

d) All HBSC should be classified either based on their risk significance or on the significance of the safety system affected;

e) The design should support personnel in the fulfilment of their responsibilities and in the performance of tasks by providing suitable and sufficient user interfaces and workspace.

Moreover, the design of the system, components, layout, Human Machine Interface (HMI) and operator working environment shall meet the human factors requirements presented in the safety case of human factors. The result of the system design will be further assessed with the Human Factors Engineering (HFE) Task Analysis. More information is presented in Chapter 15.

6.4.1.2.13 Radioactive Waste Minimisation

Waste minimisation is fundamental to radioactive waste management; reducing radioactive waste at source is an important means of waste minimisation in the UK HPR1000. Measures to control the generation of radioactive waste, in terms of both volume and radioactivity content, is considered, beginning during the design phase, and throughout the lifetime of the facility.

The control measures are generally applied in the following order of priority in line with waste hierarchy:

a) Prevent and minimise waste generation;

b) Reuse items as originally intended;

c) Recycle materials;

d) Disposal as waste.

6.4.1.2.14 Decommissioning

Decommissioning shall be considered during the design stage for the UK HPR1000. At the current stage, the general considerations of decommissioning are mentioned in Chapter 24 and mainly include:

a) Facilitating decommissioning;

b) Decommissioning strategy; and,

c) The preliminary decommissioning plan for the UK HPR1000.

During the RCP [RCS] design, the main consideration shall be given to facilitate decommissioning, and shall be fulfilled mainly by the process design, equipment design and layout design.

The related requirements and principles are presented in Decommissioning Area Safety Case (PCSR Chapter 24).

6.4.2 Design Bases

This Sub-chapter aims to provide the design bases for the RCP [RCS]. These design bases are derived from the safety requirements and are used for further equipment sizing.

Two kinds of assumption are provided as below:

a) General Assumptions

The general assumptions are mainly derived from design requirements presented in Sub-chapter 6.4.1.2. The applicable principles which may affect the equipment sizing and the relevant assumptions are demonstrated in Sub-chapter 6.4.2.1.

The applicable principles for the RCP [RCS] which may affect the equipment sizing include:

1) Safety Classification;

2) Ageing and Degradation;

3) Equipment Qualification;

4) Considerations related to the Electrical Power Grid;

5) Hazards.

b) Design Assumptions

The design assumptions are mainly derived from safety functional requirements presented in Sub-chapter 6.4.1.1. These assumptions are demonstrated in Sub-chapter 6.4.2.2.

6.4.2.1 General Assumptions

6.4.2.1.1 Safety Classification

The components constituting the pressure retaining boundary of the RCP [RCS] shall be classified as Design Provision Class 1 (B-SC1). These components include the RPV, SG (primary side), PZR, Main Coolant Line, Reactor Coolant Pumps, PSVs, SADVs and the pressure retaining boundary isolation valves as well. The secondary side of SG shall be classed as Design Provision Class 2 (B-SC2).

The valves and pipes that connect to the main circuit loop with flow limit devices and serve as the reactor coolant system pressure boundary, shall be classified as Design Provision Class 2 (B-SC2), in the case of the potential leakage can be compensated by the normal makeup method following pipe rupture.

The overpressure protection function under DBC conditions is Function Category 1 (FC1). This function is allocated to the PSVs.

The severe accident depressurisation function under DEC-B conditions is Function Category 3 (FC3). This function is allocated to the SADVs.

The safety classification for the main equipment of the RCP [RCS] is presented in Table T-6C-4. More detailed information is presented in Reference [5].

6.4.2.1.2 Ageing and Degradation

According to Chapter 2, the operational design life of the UK HPR1000 is 60 years. The main components constituting the pressure retaining boundary of the primary loop are designed for the 60-year plant operation. These components include:

a) Reactor Pressure Vessel;

b) Steam Generators;

c) Reactor Coolant Pumps;

d) Pressuriser;

e) Main Coolant Lines and Surge Line;

f) Pressuriser Safety Valves (PSVs);

g) Severe Accident Dedicated Valves (SADVs).

During system and equipment design, the ageing and degradation of equipment which is important to safety must be taken into account.

Moreover, the ageing effect of sensors such as drift shall be considered. This is applied in the monitoring control function design.

6.4.2.1.3 Equipment Qualification

All components of the RCP [RCS] performing an FC1 or FC2 safety function shall be qualified. All components of the RCP [RCS] performing an FC3 safety function required under DEC conditions shall be qualified.

6.4.2.1.4 Considerations Related to the Electrical Power Grid

Fluctuation of the electrical power grid may affect the ability of safety functions, especially the safety functions performed by active equipment such as pumps and electrical valves.

For the RCP [RCS] fluctuation of the electrical power grid affects the primary loop flowrate provided by the Reactor Coolant Pumps. The effects of flowrate change shall be further estimated in safety analysis.

The information related to the grid connection is presented in Chapter 3.

6.4.2.1.5 Hazards

The effects of internal and external hazards shall be considered in the RCP [RCS] equipment design, such as internal flooding, high energy pipe failure, and earthquakes.

These hazards may induce an environmental condition change in the compartment where the equipment is located. Moreover, safety functions such as confinement shall be performed under an earthquake event. Therefore, the equipment design shall take the effects of hazards into account to ensure the safety function ability remains.

6.4.2.2 Design Assumptions

6.4.2.2.1 Control of Reactivity

Reactor coolant inventory (best estimate value, including the water inventory contained in the PZR) of the RCP [RCS] is presented in Reference [5].

The Control Rod Drive Mechanism (CRDM) is used to provide the reactivity control function during plant normal operation and the reactor trip function during plant accident. The safety functional requirements and design principles of the CRDM is detailed in Sub-chapter 6.5.3.

The PZR spray shall provide a nominal flow rate of 2.3 kg/s to maintain uniform concentration of boric acid within the MCL and PZR, Reference [5].

The design of the RCP [RCS] shall ensure that the Reactor Coolant Pumps can be automatically tripped to prevent primary coolant over-cooling which will induce a reactivity increase in the reactor core under plant accident conditions.

The design of the RCP [RCS] shall ensure that the high pressure cooler of the Reactor Coolant Pump can be isolated in the following cases to prevent dilution risk:

a) Plant normal shutdown and primary loop pressure is lower than the RRI [CVCS] operation pressure;

b) Potential leakage or break in the tube of the high pressure cooler has been identified.

6.4.2.2.2 Removal of Heat

To ensure the heat removal function, the components of the primary loop shall be designed with a reliable integrity to ensure a coolable geometry. The component design information is presented in Sub-chapter 6.5.

During plant normal operation, the Reactor Coolant Pumps provide the necessary flow rate for core cooling. During plant shutdown, the design of the RCP [RCS] shall ensure that the Reactor Coolant Pump can be switched off in the Main Control Room (MCR) in order to limit the heat produced by the pump being transferred into the primary loop.

Under a LOCA accident, the design of the RCP [RCS] shall ensure that the Reactor Coolant Pumps can be automatically switched off in the MCR to prevent further depletion of the coolant inventory induced by pump operation.

The flywheel of the Reactor Coolant Pump shall be designed to provide adequate inertia after pump trip to remove residual heat at the early stage after reactor trip, Reference [5]. The layout of the RCP [RCS] shall be designed to ensure that the natural circuit operation can remove the residual heat of the reactor core after plant shutdown, Reference [18].

During plant power operation, the spray and heaters of the PZR are used to maintain the pressure of the RCP [RCS] to provide a suitable over-cooling margin. In a plant accident, the RCP [RCS] design shall ensure that the spray and heaters can be automatically or manually started up or shut down to support the heat removal functions based on post accidental operation.

The maximum flow rate of the spray shall be limited to prevent primary over-cooling which will induce a potential risk for the reactor core, Reference [5].

Under DEC-A conditions, the RCP [RCS] design shall ensure that the 3 PSVs can be manually opened via pilots simultaneously in the MCR. The RCP [RCS] shall ensure the PSVs can be re-closed via controlling the pilots when the F&B operation is finished.

The isolation valves (normally closed) which are installed between the RCP [RCS] and the interfacing system shall provide a reliable isolation function to maintain the primary coolant inventory. To ensure reliable isolation, double isolation is preferred to be used as engineering practice. Moreover, the RCP [RCS] design shall ensure that the isolation valves (normally opened) can be isolated automatically or manually in the MCR based on the system operation.

6.4.2.2.3 Confinement

The components of the RCP [RCS] shall be designed with a reliable integrity to ensure leaktightness of the primary loop.

PSVs perform the overpressure protection function to ensure that the maximum pressure and temperature failure limits of the RCP [RCS] will not be exceeded. Each PSV shall be designed that [5]:

a) A minimum discharge flowrate of 210 t/h saturated steam under 17.23 MPa (a) is provided. There is no maximum flowrate limitation identified for the PSVs. However, the maximum discharging flowrate fed back by the valve supplier is used in the re-estimated safety analysis;

b) The opening stroke time of each PSV shall be limited to no more than 0.1 seconds. The closing stroke time of each PSV shall be limited to no more than 1.0 seconds;

c) The dead time of valve opening shall be limited to no more than 0.5 seconds;

d) The dead time of valve closing shall be limited to no more than 5 seconds;

e) The set point for opening of PSVs shall be:

   1) The first PSV: 17.1 MPa (a);

   2) The second PSV: 17.4 MPa (a);

   3) The third PSV: 17.7 MPa (a).

SADVs provide containment overpressure protection via fast depressurisation of the

RCP [RCS] under DEC-B conditions. Each SADV train is required to provide a minimum discharge flowrate of 630 t/h saturated steam under 17.23 MPa (a), Reference [5]. There is no maximum flowrate limit identified for the SADVs. The maximum flowrate fed back by the supplier will be re-estimated in the safety analysis.

Shaft seals of Reactor Coolant Pumps provide leakage control function to limit the coolant discharged from the RCP [RCS]. Under a Station Black Out (SBO) condition, a conservative assumption is given on the seals cascade failure. In this case, the design of the shaft seal shall ensure that the leakage flow rate of the shaft seal shall be limited to no more than $0.295m^3/h$.

The RCP [RCS] design shall ensure that the potential leakage from the RPV main flange can be detected and isolated. Relevant operational information shall be provided for the operator.

The isolation valves between the RCP [RCS] and the interfacing systems (including the shaft seal leakoff line isolation valves and high pressure cooler isolation valves) provide reliable isolation function to prevent radioactive material discharge from the RCP [RCS]. There is no special time limitation relevant to the valves opening/closing. The opening/closing time of these valves fed back by the supplier will be re-estimated in the safety analysis. According to Reference [19], a bellow seal type valve is preferred to ensure leaktightness of the RCP [RCS].

6.4.2.2.4 Extra Safety Function

Important system operation parameters which indicate the safety operational status of the RCP [RCS] shall be monitored as below [5]:

a) Reactor coolant temperature (including hot leg, cold leg and average coolant temperature);

b) PZR water level which is used to indicate the reactor coolant inventory of the RCP [RCS];

c) Reactor coolant flowrate in the primary loop which is used to indicate the heat removal capacity from the reactor core;

d) Pressure of the RCP [RCS] which is used to indicate the potential risk of over-pressure or the over-cooling margin.

Important status information of components indicating the potential degradation of safety function performed by these components shall be monitored as below [5]:

a) Operational parameters relevant to the Reactor Coolant Pumps, including:

    1) Leak-off flowrate, pressure difference and temperature increase of the shaft seal;

    2) Pressure, temperature and flowrate of the cooling water for the high pressure

cooler;

3) Temperature, oil level, vibration, shaft displacement and electric current of the motor.

b) Status information relevant to the important valves, including:

1) Opening and closing status for the isolation devices of the pressure retaining boundary of the RCP [RCS], including PSV, SADVs and Reactor Coolant Pressure Boundary (RCPB) isolation valves;

2) Leaktightness monitoring, including potential leaks from the RPV main flange seal, PSVs and SADVs.

Potential hazards induced by the RCP [RCS] operation or malfunction shall be considered in the RCP [RCS] design [5]. The design of the RCP [RCS] shall ensure that:

a) Potential hazards relevant to system overpressure can be prevented:

1) PSVs performing an overpressure protection function shall ensure the integrity of the RCP [RCS];

2) The safety relief valve set on the high pressure cooler cooling piping performs an overpressure protection function due to the boiler effect after the high pressure cooler cooling line is isolated;

3) The safety relief valve set on the oil lifting subsystem of the Reactor Coolant Pumps performs an overpressure protection function for the motor to prevent the potential risk of internal fire.

b) Potential hazards relevant to hydrogen accumulation can be prevented:

1) Downstream piping of PSVs and SADVs shall be swept continually in order to prevent potential hydrogen accumulation which may induce potential risk of explosion;

2) Small de-gas from the PZR during plant normal operation shall prevent hydrogen accumulation in the upper dome of the PZR.

c) Potential hazards relevant to thermal-hydraulic phenomenon can be prevented:

1) The discharging line downstream of PSVs and SADVs shall be protected from water hammer induced by the condensation of steam discharged.

6.4.3 System Description and Operation

6.4.3.1 System Configuration

The overall schematic diagram of the RCP [RCS] is presented in Figure F-6D-1 in Appendix 6D. Moreover, the overall system configuration is presented in Appendix 2A of Reference [7].

6.4.3.2 Main Equipment

Only the main equipment forming the pressure retaining boundary of the RCP [RCS] are described in this chapter. The main equipment of the RCP [RCS] includes the RPV, (including Reactor Vessel Internals (RVI) and CRDM), MCL, SG, Reactor Coolant Pump, PZR, PSVs, SADVs and pressure retaining boundary isolation valves.

Reactor Pressure Vessel

The RPV main structure is a cylinder consisting of the closure head, the RPV body, fastening components and seals.

The equipment design is presented in Sub-chapter 6.5.1.

Reactor Pressure Vessel Internals

The RVI refers to the parts in RPV except for the fuel assembly and related components, core measuring instrument related components and irradiated sample monitoring pipes. Its functions mainly involve the mechanical integrity of the fuel assembly.

The equipment design is presented in Sub-chapter 6.5.2.

Steam Generator

The steam generator is a natural circulation U-tube heat exchanger. Its main function is to transfer the primary coolant heat to the secondary sub-cooling water and saturated steam-water mixture. The tube, tube plate and lower plenum of it are also a part of the pressure boundary.

The equipment design is presented in Sub-chapter 6.5.4.

Pressuriser (including spray and electrical heaters)

The PZR is a vertical cylindrical container with up and down spherical heads, the main function of which is to control the primary pressure when it fluctuates, to keep it within the permissible limits. It is also used to compensate for the level changes caused by unit power fluctuations. The PZR also guarantees the integrity of the pressure boundary as a primary pressure retaining boundary.

The PZR electrical heaters are the direct immersion type. The heater and spray systems are used to control the RCP [RCS] pressure. The spray flowrate is controlled by normal spray valves.

The equipment design is presented in Sub-chapter 6.5.5.

Main Coolant Line

The MCL consists of the Hot Leg, Cold Leg, Cross Leg and Surge line. It is mainly used to transport the reactor coolant, and is also a part of the primary coolant pressure boundary.

Equipment design is presented in Sub-chapter 6.5.6.

Reactor Coolant Pump

The Reactor Coolant Pump is single stage vertical shaft seal pump. The main function of it is to provide forced circulation flow for the primary loop to remove the heat (including the core fission heat in normal operation and decay heat in normal shutdown) generated by the core through the coolant. Another basic function of the Reactor Coolant Pump is to guarantee the integrity of the pressure retaining boundary.

The equipment design is presented in Sub-chapter 6.5.7.

Pressure Safety Valve

Three PSVs are planned for the RCP [RCS]. Each PSV consists of a main valve and pilots. The main function of the PSVs is to perform overpressure protection under overpressure accident conditions.

The equipment design is presented in Sub-chapter 6.5.8.

Severe Accident Dedicated Valve

Two trains of SADVs are planned for the RCP [RCS]. The SADVs are equipped with two valves installed in series with 100% functional capability for each train. The main safety function of SADVs is to depressurise the RCP [RCS] in a severe accident.

The equipment design is presented in Sub-chapter 6.5.9.

Isolation Valves

The main function of these isolation valves is to provide adequate isolating function to support the safety functions as below:

a) Preventing water inventory degradation to support the heat removal function;

b) Preventing radioactive effluent discharged from the RCP [RCS];

The information of equipment design is presented in Sub-chapter 6.5.10.

6.4.3.3 Main Layout

The RCP [RCS] is arranged in the reactor building. The general layout information of the RCP [RCS] is presented in Figures F-6D-2 and F-6D-3.

The main components of the RCP [RCS] perform functions which are important to nuclear safety. Therefore, the following requirements shall be considered during the RCP [RCS] layout arrangement design, Reference [18]:

a) The RCP [RCS] layout shall take the hazard protection principles and methodology into account;

b) The accessibility for the layout of the RCP [RCS] shall be designed to ensure that the maintenance, inspection, testing works can be carried out;

c) Radiological protection measures shall be provided to the worker who will be carrying out the works relevant to maintenance or decommissioning.

Moreover, the following issues are also considered during the RCP [RCS] layout design to improve system performance, Reference [18]:

a) The installation height of the SGs shall ensure that the inspection or maintenance work on the U-tubes of SGs can be performed by emptying the primary side of SGs and won't affect the operation of the RHR at the same time;

b) The installation height of the SGs shall ensure the natural circulation of coolant water when the reactor is shut down and all the Reactor Coolant Pumps are out of service;

c) The layout of each of the Cross-over Legs shall optimize the Net Positive Suction Head (NPSH) of the Reactor Coolant Pumps.

6.4.3.4 System Interface

Various systems are connected to the RCP [RCS] directly to support its safety functions. The main system interfaces are presented below and further design information is presented in Reference [20]:

a) Main Feedwater Flow Control System (ARE [MFFCS])

The ARE [MFFCS] is required to supply water for the SGs during plant normal operating conditions, including startup and shutdown.

b) Steam Generator Blowdown System (APG [SGBS])

The APG [SGBS] is required to provide the following supporting functions:

1) Maintain the chemical characteristics of the secondary side of the SGs and perform the wet lay-up of the SGs during periods of maintenance;

2) Transfer water between SGs under Steam Generator Tube Rupture (SGTR) accident conditions to avoid overfilling of the affected SG.

c) Emergency Feedwater System (ASG [EFWS])

The ASG [EFWS] is required to supply water for the SGs under accident conditions if the ARE [MFFCS] is unavailable.

d) Emergency Boration System (RBS [EBS])

The RBS [EBS] is required to inject borated water into the RCP [RCS] through the RIS [SIS] injection line under accident conditions.

e) Chemical and Volume Control System (RCV [CVCS])

The RCV [CVCS] is required to provide the following supporting functions:

1) Supply makeup water (borated or non-borated) to the RCP [RCS] via the charging line;

2) Control the PZR / loop level through charging/letdown balance;

3) Control the RCP [RCS] pressure during startup or shutdown (water solid operation);

4) Purifying reactor coolant and adjusting water chemistry (pH, H_2);

5) Provide auxiliary spray for the PZR if normal spray is unavailable;

6) Provide shaft seal injection water and collecting the seal leakage.

f) Nuclear Sampling System (REN [NSS])

The REN [NSS] is required to take samples from the RCP [RCS] and SG secondary side.

g) Safety Injection System (RIS [SIS])

The RIS [SIS] is required to provide the following functions for the RCP [RCS]:

1) Removal of core residual heat when operated in Residual Heat Removal (RHR) modes;

2) Provide the safety injection function under accident conditions;

3) Provide cold overpressure to ensure the integrity of primary loop;

4) Provide injection (via the accumulator) to prevent cavitation of Reactor Coolant Pumps induced by the pressure fluctuation of the RCP [RCS].

h) Nuclear Island Vent and Drain System (RPE [VDS])

The RPE [VDS] is required to provide the following supporting function:

1) Collect and condense permanent degassing of the PZR through degassing line during normal operation;

2) Provide vacuuming extraction before the RCP [RCS] startup;

3) Collect potential leakage if the RPV flange seal fails;

4) Drain the cross leg and the Pressuriser Relief Tank (PRT);

5) Collect low-pressure leak-off and flushing water of the reactor coolant pumps;

6) Cool and depressurise the PRT.

i) Component Cooling Water System (RRI [CCWS])

The RRI [CCWS] is required to supply cooling water for the motor air coolers, oil coolers and high pressure cooler assembly of the Reactor Coolant Pumps.

j) Nitrogen Distribution System (SGN [NDS])

The SGN [NDS] is required to provide the following supporting functions:

1) Filling of the PZR under nitrogen atmosphere during shutdown and draining;

2) Primary nitrogen sweeping before opening the RPV upper head;

3) Driving compressed nitrogen for the standstill seal system of the Reactor Coolant Pump.

k) Main Steam System (VVP [MSS])

The VVP [MSS] is required to transfer steam produced from the SGs.

l) Atmospheric Steam Dump System (VDA [ASDS])

The VDA [ASDS] is required to remove residual heat under accident conditions or if the GCT [TBS] is unavailable.

6.4.3.5 System Instrumentation and Control

Instrumentation is designed to detect any degradation of the capability for core cooling or any deterioration of components important to safety.

Important operating parameters for heat transport are detected and provide information to the operators, including the pressure, temperature, and water level of the RCP [RCS].

Leaks of reactor coolant are also monitored by I&C system design to indicate the degradation of the RCPB, both the leakage of primary side and the leak to the secondary side from the tubes of the SGs.

The RCP [RCS] and its supporting systems provide several control functions to keep the plant operating within the safety limit. These control functions are as follows:

a) RCP [RCS] pressure control function;

b) PZR level control function;

c) RCP [RCS] Loop level control function;

d) SG level control function.

The Instrumentation and Control (I&C) function of the RCP [RCS] is presented in Reference [21]. The information of the UK HPR1000 I&C systems design is presented in Chapter 8.

6.4.3.6 System Operation

6.4.3.6.1 Plant Normal Condition

Table T-6C-3 presents the basic parameters of the RCP [RCS]. Figure F-6D-4 presents the temperature of the RPV inlet and outlet and the average temperature consequently, depending on the power. The parameters presented in the figure are based on the Best Estimate (BE) flow rate of the reactor coolant with three Reactor Coolant Pumps in normal operation. The pressure of the SGs changes with different power loads and is shown in Figure F-6D-5.

The standard operating states of the plant as well as the general parameters relevant to the operation of the RCP [RCS] are presented in Reference [22] as shown below:

a) Reactor in Power Mode (RP)

In this mode:

1) The reactor is critical or approaching criticality;

2) The RCP [RCS] is closed and filled, the PZR is biphasic;

3) Coolant average temperature: 295°C - 307°C;

4) The RCP [RCS] pressure: 15.5 MPa (a).

b) Normal Shutdown with Steam Generators Mode (NS/SG)

In this mode:

1) The reactor is subcritical;

2) The RCP [RCS] is closed and filled, the PZR is biphasic;

3) The heat in RCP [RCS] is removed by the steam generators;

4) Coolant average temperature: 135°C - 295°C;

5) The RCP [RCS] pressure: 2.4 MPa (a) - 15.5 MPa (a).

c) Normal Shutdown with RIS-RHR Mode (NS/RIS-RHR)

In this mode:

1) The RCP [RCS] is closed or, open and non-pressurised, and filled, the PZR is monophasic or biphasic;

2) Heat in the RCP [RCS] is removed by the RIS [SIS];

3) Coolant average temperature: 10°C - 140°C;

4) The RCP [RCS] pressure: less than or equal to 3.2 MPa (a).

d) Maintenance Cold Shutdown Mode (MCS)

In this mode:

1) The RCP [RCS] is open and non-pressurised;

2) The RCP [RCS] level is greater than or equal to the lowest level of operation interval of RIS-RHR, and less than the level when the reactor cavity is filled;

3) Heat in RCP [RCS] is removed by RIS [SIS];

4) Coolant average temperature: 10°C - 60°C;

5) The RCP [RCS] pressure is atmospheric.

e) Refuelling Cold Shutdown Mode (RCS)

In this mode:

1) The reactor cavity is filled;

2) At least one fuel assembly is in the reactor building;

3) Heat in the RCP [RCS] is removed by RIS [SIS];

4) Coolant average temperature: 10°C - 60°C.

f) Reactor Complete Discharged Mode (RCD)

In this mode, the reactor building is without any fuel assembly.

Reactor Coolant Pumps

During most of normal plant operation, 3 Reactor Coolant Pumps are in operation to provide adequate flowrate for core cooling.

Shaft seals of the Reactor Coolant Pumps provide a controlled leak from the RCP [RCS]. The injection water of the pump shaft seal is provided by the RCV [CVCS], and the seal leakoff is collected by the RCV [CVCS]. If the RCV [CVCS] malfunctions which induces the loss of seal injection water, the high pressure cooler of the Reactor Coolant Pump, which is provided cooling water by the RRI [CVWS], will be used to cool the fluid coming from the RCP [RCS] to protect the shaft seal assembly, Reference [21].

The upper and lower oil-lubricated radial and double-thrust bearings of the pump motor are provided with cooling water by the RRI [CCWS]. During pump operation, oil level of the motor is monitored to detect any potential function degradation. The jacking oil system is put into service before startup or shutdown of the Reactor Coolant Pump, Reference [21].

Pressuriser (including spray and heaters)

During normal plant operation, including normal startup and shutdown, spray and heaters are used to control the RCP [RCS] pressure when the PZR is vapour-liquid two phase, Reference [21]. This is to maintain a suitable over-cooling margin for core cooling.

The plant load change will induce a reactor coolant average temperature change resulting in a coolant volume change:

a) To prevent RCP [RCS] pressure over-increase, the reactor coolant sprays into the steam space and condenses a portion of steam, and then reduces the pressure of RCP [RCS];

b) To prevent RCP [RCS] pressure over-decrease, the heaters are automatically started, heating the remaining water in the PZR, therefore limiting pressure reduction.

During primary water solid operation stage, the pressure of the RCP [RCS] is controlled by the RCV [CVCS].

During normal power operation, the water level of the PZR is controlled via RCV [CVCS] based on the power load.

Pressure relief system

The pressure relief system includes pressure relief devices (i.e. PSVs), discharge piping and the Pressuriser Relief Tank (PRT), Reference [23].

During normal plant operation, the pressure relief system is on standby. The pipes downstream the PSVs and SADVs are continually swept by nitrogen to prevent any potential hydrogen accumulation, Reference [21].

RPV flange leaktightness monitoring

Between the inner and outer C-ring of the RPV, a leaktightness monitoring subsystem is used to provide a continual monitoring function to detect any potential leakage of the RPV main flange, Reference [21].

6.4.3.6.2 Plant Fault or Accident Condition

The safety analyses are demonstrated and analysed for the fault or accident condition of the RCP [RCS]. More information is presented in Reference [21].

6.4.4 Design Substantiation

This Sub-chapter provides information relevant to the RCP [RCS] design in order to demonstrate that the safety requirements (including safety functional requirements as well as design requirements) are fulfilled.

6.4.4.1 Compliance with Safety Fundamental Requirements

Information regarding RCP [RCS] system design is presented in References [20], [21] and [23]. At the current stage no further safety functional requirements are identified for the RCP [RCS]. The system configuration and the capability of the components are complied with by these functional requirements. The details of the RCP [RCS] design, especially the design parameters of the main equipment, are provided in Chapters 12 and 13.

6.4.4.1.1 Control of Reactivity

The design information of CRDM is presented in Sub-chapter 6.5.3.

The PZR spray has been designed so that it can provide a flow rate of 2.3 kg/s to ensure uniform concentration of boric acid within the MCL and PZR, Reference [20].

The design of the I&C control function of the RCP [RCS] ensures that:

a) The Reactor Coolant Pumps can be automatically tripped under accident conditions to prevent primary loop over-cooling, Reference [21];

b) The cooling line of the high pressure cooler can be automatically isolated based on system operation or a potential tube break in the high pressure cooler, Reference [21].

6.4.4.1.2 Removal of Heat

The Reactor Coolant Pump is designed to provide an adequate flow rate for reactor core cooling.

The inertia provided by the flywheel design of the Reactor Coolant Pumps, as well as the RCP [RCS] layout design for natural circulation, have been estimated by safety analysis (see Chapter 12).

The design of the I&C control function of the RCP [RCS] ensures that:

a) The Reactor Coolant Pumps can be automatically tripped under LOCA accident conditions to prevent coolant inventory depletion induced by pump operation, Reference [21];

b) The cooling line of the high pressure cooler can be automatically isolated in the MCR based on system operation or a potential break in the high pressure cooler, Reference [21].

The maximum flowrate of spray has been designed as no more than 64kg/s, Reference [20]. The design of the I&C control functions of the RCP [RCS] ensures that pressure can be properly controlled during plant operation, Reference [21].

The I&C control function design of the PSVs ensures that the 3 valves can be manually opened from the MCR simultaneously via electromagnetic pilots, Reference [21]. During F&B operations, the I&C design ensures that the PSVs can be kept open until the operator closes the valves based on the plant operation procedure.

Double isolation is widely used in the RCP [RCS] to ensure reliable isolation, Reference [23]. Moreover, the I&C design of the RCP [RCS] ensures that the isolation valves can be closed automatically based on I&C signal or manually by the operator in the MCR based on the demand of system operation, Reference [21].

6.4.4.1.3 Confinement

The components of the RCP [RCS] are designed to ensure leaktightness of the primary loop. Analysis of the RCP [RCS] main components relevant to structural integrity is presented in the safety case of Chapter 17.

The design parameters of the PSVs are presented in Table T-6C-15, Reference [20]. These parameters are used in the safety analysis and fulfil the requirements of the overpressure protection function.

The design parameters of the SADVs are presented in Table T-6C-16, Reference [20]. These parameters are used in the safety analysis and fulfil the requirements of fast depressurisation of the RCP [RCS] under DEC-B condition.

Shaft seals are designed to ensure leakage control function during normal plant operation, Reference [20]. Under SBO conditions, shaft seals are designed to have the ability to limit the leakage flowrate to no more than $0.295m^3/h$. The design results fulfil the safety functional requirements and are estimated in Chapter 13.

The RCP [RCS] design ensures the potential leakage from the RPV main flange can be detected and isolated. A leaktightness monitoring subsystem is designed to fulfil this safety functional requirement. The temperature and pressure information indicates the potential leakage, and relevant alarm and automatic isolation control functions are provided to the operator in the MCR, Reference [21].

6.4.4.1.4 Extra Safety Function

The I&C of the RCP [RCS] is designed to ensure that the important system operation information as well as important status information of components can be indicated to the operator, References [21] and [23].

Moreover, the results of the RCP [RCS] design presented in References [18], [20], [21] and [23] show that the extra safety functional requirements mentioned in Sub-chapter 6.4.1.1.4 have been substantiated.

6.4.4.2 Compliance with Design Requirements

6.4.4.2.1 Safety Classification

The principles concerning safety classification are summarised in Sub-chapter 4.4 and more detailed information is presented in References [3] and [8].

The safety classification for the main equipment of the RCP [RCS] is demonstrated in Sub-chapter 6.4.2.1.1 and presented in Table T-6C-4. More detailed information is presented in Reference [5].

The RCP [RCS] design complies with these principles.

6.4.4.2.2 Engineering Design Requirement

The Reliability Design of SSC

a) Single Failure Criterion (SFC)

The principles of SFC are presented in Sub-chapter 4.4 and Table T-7.1-1 in Reference [8].

The SFC is considered for the FC1 opening function of PSVs which perform the overpressure protection function. Therefore, 3 PSVs are designed to ensure the reliability of the safety function.

The SFC is considered for FC2 isolation function. Therefore, the valves are set in series on the injection line and leakoff line for the high pressure cooler of Reactor Coolant Pump.

More information relevant to the design of the RCP [RCS] is presented in References [20], [21] and [23].

b) Independence

According to the principles presented in Sub-chapter 4.4, and the high level principle of independence between levels of DiD, the following principles are applied in the RCP [RCS] design to achieve system reliability and tolerance to faults:

1) Independence between the three loops of the RCP [RCS] is maintained as far as reasonably practicable to prevent common cause failure;

2) Independence between components of different safety categories of the RCP [RCS] is maintained as far as reasonably practicable to prevent impact on a component of higher safety category from an item of a lower safety category;

3) Physical separation is applied in the layout design of the RCP [RCS] as far as reasonably practicable, to reduce the potential of common cause failure due to a localised initiating event.

c) Diversity

The principles concerning Diversity are presented in Sub-chapter 4.4. The design of the RCP [RCS] complies with these principles.

Two kinds of sensor are provided to the high pressure cooler cooling water line for each of the Reactor Coolant Pumps. One is a pressure sensor and the other a temperature sensor. In the event of a tube break in the high pressure cooler, the pipe line can be isolated reliably on the basis of both temperature and pressure, thus preventing radioactive material from being discharged into the environment directly.

Overpressure protection under cold shutdown conditions is provided both by the RIS [RHR] and the RCP [RCS]. The PSVs serve as back up for the RIS [SIS] safety valve to perform the overpressure protection functions.

More information is presented in References [20], [21] and [23].

d) Fail-Safe

The fail-safe design has been considered in the RCP [RCS] design. The valves fail-safe position shall be analysed and defined during design, and the design requirement will be provided to the equipment vender selected.

The solenoid pilots of the PSVs are designed to prevent spurious opening of the main valves if the electrical power supply is lost.

The CRDM design ensures that the control rod can be inserted via gravity if electrical power is lost. Thus, the reactor can be shut down safely.

The system configuration and the component design are presented in References [20], [21] and [23].

e) Ageing and Degradation

The target service life of the main equipment of the RCP [RCS] is identified in Sub-chapter 6.4.2.1.2.

The performance of equipment is guaranteed through life EMIT and by monitoring during normal operation. Thus, it can be ensured that ageing effects will not compromise safety performance. The detailed design arrangements around EMIT and equipment monitoring is presented in Reference [23].

The system layout design can ensure the accessibility and requirement for safety equipment in-service inspection and periodic tests including the necessary NDT are met. This also includes the requirements of emergency and scheduled maintenance on the SSC. Detailed layout information is presented in Reference [18].

<u>Autonomy</u>

a) Autonomy with respect to Operators

The design principles relevant to autonomy with respect to operators are listed in Reference [3]. Applicable principles for the RCP [RCS] are listed below:

1) The numerical targets of DBC-2, DBC-3, DBC-4 and DEC-A can be met without operator action from the MCR in less than 30 minutes from the first significant signal;

2) The numerical targets of DBC-2, DBC-3, DBC-4 and DEC-A can be met without action outside of the MCR in less than 1 hour from the first significant

signal.

Generally, these principles are mainly fulfilled by the control function design mentioned above in the "*Human Factors*" section. More information is presented in References [20], [21] and [23].

b) Autonomy with respect to the Heat Sink

The design principles relevant to the autonomy in respect to the heat sink are listed in Reference [3]. These principles are not applicable for the RCP [RCS] design.

However, the heat sink design shall ensure that the primary heat can be removed in order to maintain reactor core cooling. The thermal load of the RCP [RCS] is one of the design inputs for the heat sink systems.

c) Autonomy with respect to Power Supply Systems

Though the principles relevant to the electrical power supply are not applicable for the RCP [RCS], the design of the RCP [RCS] (including the system control function design) must consider the general principles of electrical power distribution which are described in Chapter 9.

Considering Loss of Offsite Power (LOOP) and Station Black Out (SBO) accidents, the components that perform safety functions are equipped with emergency power supplies as detailed below:

1) Parts of the electrical heaters are connected to the Emergency Diesel Generator (EDG);

2) The solenoid pilots of the PSVs are connected to the diesel generators (e.g. EDG, SBO diesel generator);

3) The RCPB isolation valves are connected to the EDG, SBO diesel generators and a 2 hours un-interrupted battery;

4) The two trains of SADVs are connected to the EDG, SBO diesel generators and a 24 hours un-interrupted battery.

<u>Other design requirements</u>

a) Prevention of Harmful Interactions of Systems Important to Safety

According to Reference [3], protection of interfacing systems shall be considered in the RCP [RCS] design.

The RCP [RCS] operates at high pressure and temperature normally. Various systems are connected to the RCP [RCS] to support the safety function or normal operational function. According to Reference [3], provision shall be made to prevent the design pressure of the system operating at the lower pressure from being exceeded. This is mainly achieved by the design measures stated below, Reference [23]:

1) Adequate isolation (e.g. using double isolating valves or a physical disconnect) between the RCP [RCS] and interfacing systems;

2) Appropriate control function design to protect the interfacing system from overpressure via automatic isolation;

3) Safety relief devices such as safety valve (PSVs, high pressure cooler cooling line safety valve) and rupture disks (set on the pressuriser relief tank) to prevent overpressure.

b) Considerations Related to the Electrical Power Grid

According to Reference [3], the functionality of items important to safety of the nuclear power plant shall not be compromised by disturbances in the electrical power grid.

For the RCP [RCS], fluctuation of the electrical power grid mainly affects the functional capability of the Reactor Coolant Pumps which provide the adequate flow rate for core cooling. In the reference design, the requirement of overcoming the fluctuation of the electrical power grid has been considered and the design requirement is presented in the equipment specification for the pump supplier. The design result and feedback from the supplier has been included in the safety analysis.

6.4.4.2.3 Equipment Qualification

The seismic classification for the main equipment of the RCP [RCS] is presented in Table T-6C-4. More information is presented in Reference [5].

Moreover, a safety case supporting document ‘qualification schedule’ (Reference [24]) has been produced to capture important information supporting the relevant engineering design and safe demonstration. The document:

a) Clearly presents the SSCs which are important to safety, with their associated safety functions, specifies their service conditions and defines the key performance requirements which shall be qualified;

b) Provides the link between the equipment qualification requirements identified in the safety case and the design of SSCs.

6.4.4.2.4 Protection against Internal and External hazards

<u>Protection against Internal Hazards</u>

The protection against internal hazards mainly depends upon the design of buildings, rooms, fire compartments and anti-flooding compartments associated with the RCP [RCS]. The specific protection design is presented in Reference [18]. The evaluation of the design is presented in Chapter 19.

Protection against External Hazards

a) Earthquakes

The components classified as B-SC1 and B-SC2 or components performing the FC1 and FC2 safety functions are designed and classified as SSE1. The components performing the FC3 safety functions and contributing to protection and mitigation in DECs are also designed and classified as SSE1. These components shall be qualified to ensure their safety functions, Reference [5].

Besides the system design, protection from earthquakes is also considered in the reactor building civil design. The information relevant to the reactor building is presented in Chapter 16.

b) Other External Hazards

The RCP [RCS] protects against external disasters mainly through the building design. The specific protection design is presented in Reference [18]. The evaluation of the design is presented in Chapter 18.

6.4.4.2.5 Commissioning

Initial testing (e.g. Factory Acceptance Test (FAT)) of components before delivery to site shall be undertaken to ensure that the safety functions of these components can be properly performed. The design requirements will be provided to the equipment vendor in form of a technical specification.

After the components have been delivered to the site, a structured systematic and progressive test programme will be implemented for the RCP [RCS] to confirm the safety functional performance as well as operational performance of the RCP [RCS].

The commissioning arrangements of the UK HPR1000 will be mainly adapted from those developed for the Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)) (examples of commissioning of the RCP [RCS] in the HPR1000 (FCG3) are presented in Preliminary Safety Report (PSR) Chapter 6).

Further detailed site specific arrangements for the UK HPR1000 commissioning activities, in addition to those described in Chapter 30, will be presented during the nuclear site licensing phase in conjunction with the site license.

During GDA the system commissioning programme of the UK HPR1000 RCP [RCS] is still under development for the future operator. The methodology of the system commissioning programme is presented in Reference [25]. The preliminary system commissioning programme is presented in Reference [21].

6.4.4.2.6 Examination, Inspection, Maintenance and Testing

According to Chapter 2, the operational design life of the UK HPR1000 is 60 years. The main components constituting the pressure retaining boundary of the primary loop are designed for the 60 years plant operation. These components mainly include:

a) Reactor Pressure Vessel;

b) Steam Generators;

c) Reactor Coolant Pumps;

d) Pressuriser;

e) Main Coolant Lines and Surge Line;

f) Pressuriser Safety Valves;

g) Severe Accident Dedicated Valves.

Based on engineering experience and operational feedback, the commonly used isolation valves constituting the pressure retaining boundary do not require a strict 60 years' design life since the replacement of these valves will not affect the safety and operational performance of the RCP [RCS] prominently.

According to Chapter 31, the design requirement is that the mechanical components, are designed, manufactured and assembled so that all of the welds can be inspected. The components include RPV, CRDM, SG, PZR, reactor coolant pipework, and reactor coolant pumps, which require ISI.

In the reference design, the layout design of the RCP [RCS] has been substantiated to ensure that the accessibility and the requirements for safety equipment ISI and periodic tests are met.

For the UK HPR1000, the layout design takes UK context into account such as the height of workers, and other relevant requirements derived from UK RGPs. The layout design in the detailed design stage will consider and capture the following requirements:

a) The need for SSC replacement;

b) The accessibility and requirement for safety equipment in-service inspection and periodic tests;

c) The requirements of emergency and scheduled maintenance on the SSC over the life span of the plant.

Some examples of the RCP [RCS] periodic tests for the UK HRP1000 include:

a) RCP [RCS] leak rate test;

b) Operability of PSVs;

c) Operability of SADVs, and;

d) Operability of isolation valves.

System periodic test completeness of the RCP [RCS] for the UK HPR1000 is preliminary developed during GDA. A description of RCP [RCS] system inspection is presented in Reference [21].

6.4.4.2.7 Special Thermal-Hydraulic Phenomena

As mentioned in Sub-chapter 6.4.1.2.7, the hydraulic phenomena listed below are considered in the RCP [RCS] design.

<u>Phenomenon regarding the dead leg</u>

The RCP [RCS] is a high temperature fluid system. The pipelines between the double isolation devices are carefully protected from damage induced by the dead leg.

The pipe layout design and the insulation design ensure that the water temperature before the first SADV maintains a low temperature (the same with the environmental condition). In this way, the SADVs are protected from being damaged by the dead leg phenomenon. Therefore, the risk induced by dead leg phenomenon is eliminated.

<u>Phenomenon regarding the hot water and cold water mixing</u>

During plant operation, reactor coolant is continually purified and cleaned-up by RCV [CVCS], and then flows back to the RCP [RCS] via a charging function. In order to prevent hot-cold water mixing which may challenge the leaktightness of the RCP [RCS], design measures are provided as detailed below:

a) The heat exchanger configuration of the RCV [CVCS] ensures that the temperature difference between charging flow and primary loop is limited to as low as reasonably practicable;

b) The hot-cold water mixing phenomenon is taken into account during MCLs design and engineering practice is provided to enhance the leaktightness of the MCLs.

Therefore, the risk induced by hot-cold water mixing is reduced.

<u>Phenomenon regarding water thermal stratification</u>

During plant operation under steady-state conditions, thermal stratification may occur in the pipelines which contain high temperature water due to heat being lost. The typical example is the SL.

In order to avoid the thermal stratification phenomenon, design measures are provided as detailed below:

a) The layout design of the SL ensures that the thermal stratification phenomenon is limited;

b) A continual small flowrate spray is provided to further limit the thermal stratification phenomenon occurring on the SL, Reference [21].

Therefore, the risk induced by thermal stratification is reduced.

Phenomenon regarding the water hammers

During RCP [RCS] design in early stage of the reference plant, a potential water hammer phenomenon is identified for the pressure discharging system downstream of the PSVs. This phenomenon is induced by the condensation of steam discharged by PSVs during periodic test or under accident conditions.

In order to eliminate the water hammer phenomenon, a pressure balance device is installed to break the vacuum during steam discharging, Reference [23]. Mitigation in the form of piping layout optimisation is also provided to minimise the effect of potential water hammer (bullet effect) due to the quick opening of the PSVs.

Phenomenon regarding the boiler effect

During plant normal operation, the RCP [RCS] contains reactor coolant with a high temperature and pressure. The boiler effect induced by the high temperature of reactor coolant may result in the loss of the opening ability of gate valves. This failure may result in a potential risk under the conditions in which the valves are required to be opened to perform safety functions.

In order to eliminate the boiler effect, small by-passes are provided for the gate valves of the RCP [RCS] (i.e. the first isolation valve of the SADVs). In this way, the risk to safety induced by the boiler effect is eliminated.

6.4.4.2.8 Material Selection

Based on the principles of material selection mentioned in Sub-chapter 6.4.1.2.8, and also from the engineering experience, the material selected for the each component is presented in Sub-chapter 6.5.

6.4.4.2.9 Insulation

The principles of insulation design are mentioned in Sub-chapter 6.4.1.2.9. The insulation design for each component is presented in Sub-chapter 6.5.

At the current stage, the insulation of the main RCP [RCS] equipment is consistent with the reference design. During step 2 of GDA, lessons learned from UK context shows that the Reflective Metallic Insulation (RMI) is preferred to be used as much as possible in the primary loop. Attention is paid to reducing the debris in order to prevent the filter of the IRWST from potentially blocking under accident conditions (e.g. under a LOCA accident).

In order to ensure the risk to nuclear safety is maintained as low as reasonably practicable, an ALARP analysis for insulation used in primary loop has been conducted, the detailed information is presented in Reference [6].

6.4.4.2.10 Conventional Safety

The conventional health and safety risks to workers and the public that may arise during the construction, commissioning, operation, maintenance, and decommissioning of RCP [RCS] are identified and assessed, and the corresponding design mitigations are developed to eliminate, reduce, isolate and control them so far as is reasonably practicable using the risk management methodology detailed in PCSR Chapter 25. And the processes are recorded by RCP [RCS] Conventional Health and Safety Design Risk Register, Reference [26].

6.4.4.2.11 Human Factors

The design requirements relevant to human factors are mentioned in Chapters 4. The principles and methodology are mentioned in Chapter 15.

For the RCP [RCS] design, key consideration is given to prevent human error. This is achieved by the following design measures:

a) Allocating the safety functions to manual activity and automatic control appropriately;

b) Providing necessary information to the operator.

During plant normal operation, the PZR water level, primary pressure and reactor coolant average temperature of the RCP [RCS] are controlled automatically. Therefore, the dependence on human action is reduced.

Under accident conditions, the RCPB isolation function must be performed as quickly as possible to prevent reactor coolant from discharging in order to maintain the heat removal function. Thus, this isolation function is designed as an automatic control function to prevent human error.

The system design, as well as the control function design of the RCP [RCS], does not require short term operator intervention. It can be claimed that no operator action within 30 minutes after the initial event is required.

Moreover, the status indicators (e.g. stem limit switch) are set for the manual valves which may induce potential reactor coolant leakage if they are left in wrong opening or closing position after maintenance. The positions of valves are indicated in the MCR and alarms are designed to inform the operator. In this way, the human error can be identified and corrected.

CGN carried out human action analysis according to the safety / duty functions performed by the SSC during GDA. The outcome is presented in the Reference [27].

Additionally, CGN developed the local area HFE guidelines, Reference [28]. Then CGN carried out review work related to the local area HMI and workplace, Reference

[29]. Moreover, CGN carried out the baseline human factors assessment, Reference [30]. More detailed information is presented in the safety case of Human Factors.

6.4.4.2.12 Radioactive Waste Minimisation

Design features and operational procedures for waste generation and control include following aspects:

a) The appropriate selection of processes, design options, materials, and SSC for the facility;

b) The use of effective and reliable techniques and equipment;

c) Recycling and re-use of material to minimise generation of radioactive waste;

d) The containment of radioactive materials so as to maintain RCP [RCS] integrity to minimise radioactive leakage;

e) Provision of equipment to prevent the spread of radioactive contamination by leakage;

f) Adequate zoning to prevent the spread of contamination.

Detailed substantiation related to the system design is presented in the relevant ALARP demonstration report of PCSR Chapter 23.

6.4.4.2.13 Decommissioning

The principles and methodology are presented in Chapter 24 (Decommissioning). The design of the RCP [RCS] takes these principles and the methodology into account. The consistency evaluation is presented in Reference [31].

6.4.5 Functional Diagram

The simplified system functional diagrams are presented in Appendix 6B. The detailed system functional diagrams are presented in Reference [23].

6.5 Description of Main Components

6.5.1 Reactor Pressure Vessel

6.5.1.1 Safety Functional Requirements

During plant normal operation and under accident conditions, the main safety functional requirements of the RPV are decoupled from the safety functional requirements of the RCP [RCS] and are presented below:

a) Supporting and aligning the CRDM to support the control of reactivity;

b) Containing the core, supporting and aligning the RVI, and forming a coolant flow channel with the RVI to ensure the transfer of heat from the core;

c) Forming part of RCPB (second barrier), containing high pressure, high temperature primary coolant to ensure the confinement of radioactive material throughout its design life;

d) Providing access to the core measuring instrumentation to support extra functions of the RCP [RCS].

6.5.1.2 Description

The RPV is the highest reliability pressure boundary to contain the reactor core, core support structure and light water coolant, and provides support and position for the CRDM, RVI and core measuring instrumentation. It consists of the closure head, the RPV body, fastening components and seals. The main design parameters of the RPV are shown in Table T-6C-5; the RPV structural diagram is shown in Figure F-6D-6.

a) Closure Head

The RPV closure head consists of a single forging of flanged hemispherical upper dome, CRDM adapters, measuring instrumentation adapters, vent pipe and lugs.

There are 58 stud holes on the head flange. The lower face of head flange is clad and the cladding is grooved to form the recess or housing for the two C-sealing rings. The upper head is welded with 68 CRDM adapters, 12 measuring instrumentation adapters and a vent pipe.

The CRDM and measuring instrumentation adapters consist of adapter flanges and adapter sleeves. The adapter sleeve is made of Inconel 690 and is connected to the inside of the head assembly through seal welds.

The head assembly of the reactor pressure vessel is aligned and positioned with the reactor pressure vessel internals by relying on a set of four-in-one alignment pins installed on the head flange and vessel flange.

The entire inner surface of the reactor pressure vessel is cladded with stainless steel.

b) RPV Body

The RPV body mainly consists of the flange-nozzle shell, core shell, transition ring, lower head, inlet and outlet nozzles, and nozzle safe ends.

The flange-nozzle shell has a total of 58 threaded holes to install studs for head sealing. Three of the threaded holes can be installed with a guide rod for head alignment and positioning. A seal ledge is welded on the outside of the flange for fixing the cavity seal and preventing the reactor coolant in the reactor pool from entering the cavity during fuel loading and unloading. There is a support ledge on the inside of the flange for supporting the barrel, which has four rectangular slots for supporting and positioning of the RVI.

There are 3 pairs of inlet and outlet nozzles in the circumferential direction of the flange-nozzle shell. The angle between the inlet and outlet nozzle for each loop is 50°, and 3 loops are distributed symmetrically at an angle of 120° in the circumferential direction of the RPV. There is a ledge on the inside of the outlet nozzle, which fits with the outlet nozzle of the barrel. The stainless steel safe ends are welded on the out end of the inlet and outlet nozzles, which are made of a material similar to the MCL to ensure the quality of the weld between the MCL and RPV. Each inlet and outlet nozzle is forged with a supporting pad at their bottom, so as to place the RPV on the support structure.

c) Fastening Components and Seals

The fastening assembly of the RPV is composed of 58 sets of studs, nuts and washers, and used to guarantee the seal of the RPV and the integrity of the pressure boundary. The bottom of the stud fits with the vessel flange threaded hole, and the top applies a pre-tightening load on the head flange through the nut and washer.

The sealing assembly consists of two C-sealing rings and their fastening devices. The sealing between the head and vessel assemblies can be achieved through the C-sealing rings installed inside the two ring grooves on the lower surface of the head flange.

d) RPV support

RPV support is used to support the RPV and bear the weight of the reactor body and its related equipment and media, as well as the loads produced by supported components under various conditions, while transferring them to the reactor pit concrete. Under normal operating conditions, RPV radial movement caused by temperature and pressure expansion is allowed. Under earthquake or the accident conditions, RPV support can limit the lateral movement of the RPV. The RPV support structure diagram is shown in Figure F-6D-7.

e) Equipment Insulation

The main functional requirements of RPV insulation are as follows:

1) During normal operation conditions, the insulation is used to reduce the heat loss of the reactor;

2) During severe accident conditions, as an important part of the cavity injection and cooling system, it shall form a specified annular passage combined with the RPV outer surface to remove heat from the core.

RPV insulation is a reflective metallic type, constructed of grade Z6CN18.09 or equivalent stainless steel. The structure of RPV insulation is shown in Figure F-6D-8.

6.5.1.3 Design Principles and Codes

The RPV is designed according to internationally recognised codes, standards and takes into account operating experience. The RPV is of a similar design to typical Pressurised Water Reactor (PWR) RPV designs which use similar proven materials and manufacturing processes.

The design of the RPV considers the following basic principles:

a) The structure of the RPV meets the safety functional requirements;

b) Using large forgings as far as possible to reduce the number of welds in the pressure boundary;

c) Facilitate manufacturing, inspection and maintenance.

The structural design of the RPV complies with the provisions of RCC-M Subsection B, and the material, NDT, welding and manufacturing respectively conform to the requirements of RCC-M Subsection M, Subsection MC, Subsection S and Subsection F. Pre-service inspection and in-service inspection of the RPV components follow the stipulations of the RSE-M rule.

6.5.1.4 Classification

According to the safety categorisation and classification presented in Reference [8] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of the RPV is presented in T-6C-4.

6.5.1.5 Materials, Manufacture and Inspection

The main material of the RPV is 16MND5, and the surface cladding is austenitic stainless steel. The nozzle safe ends are manufactured with austenitic stainless steel forgings Z2CND18-12 (nitrogen controlled). For the adapter sleeves and radial support keys, the specified grade is NC 30 Fe. The procurement and manufacture of the material meets the requirements of RCC-M. The selected materials were considered fully using engineering application experience and degradation mechanisms. The degradation mechanisms of the RPV materials mainly include irradiation embrittlement, thermal ageing, temper embrittlement, fatigue, corrosion and wear.

The filler material is qualified in accordance with the requirements of RCC-M S5000. The acceptance test is performed on each lot of filler material in accordance with the requirements of RCC-M S2000.

For the low alloy steel welds, filler material according to RCC-M S2820 and S2830 are used, in addition, the Reference Nil Ductility Transition Temperature (RT_{NDT}) and the Upper Shelf Energy (USE) are required as indicated below:

a) RT_{NDT}: no more than -30°C;

b) USE: no less than 130J.

Furthermore, the KV-T transition curve is required for RPV weld metal.

For dissimilar metal welds or buttering, filler metal ERNiCrFe-7/E NiCrFe-7 according to RCC-M S2981 or RCC-M S2986 is used.

For corrosion resistant cladding, filler material 309L according to RCC-M S2930 or RCC-M S2970 is used, or 308L according to RCC-M S2920 or RCC-M S2960 is used.

Stainless steel and nickel base alloy filler material in contact with the primary coolant shall have a cobalt content of no more than 0.06%.

The forgings of the RPV components are manufactured according to the certificated processing procedures qualified under the requirements of RCC-M M140. The chemical element content of Copper, Phosphorus, Sulphur and Cobalt are more strictly controlled than in the RCC-M procurement specification. The chemical element content of Vanadium and Aluminium are controlled in the same as RCC-M procurement specification. The circumferential welds of the RPV assembly all adopt the narrow gap full penetration welding process with low alloy material.

The non-destructive inspection tests of the RPV components during fabrication comply with the requirements of RCC-M Subsection M and Subsection MC.

The PSIs and ISIs are subject to the requirements of RSE-M code involving welds and positions to be inspected, range of such positions and required methods for inspection. The non-destructive inspection equipment, procedure and personnel have competent technical characteristics and ensure defect detectability under actual inspection conditions to ensure effectiveness and reliability of non-destructive inspection techniques and the recording of accurate results.

6.5.1.6 Structural Integrity

The structural integrity of the RPV is demonstrated in Reference [33] in the form of Claims, Arguments, Evidence (CAE) from a number of aspects, which mainly include applicable design codes and standards, loading conditions, design analysis, selected material, manufacture, manufacturing inspection, operation, maintenance and defect tolerance assessments.

6.5.2 Reactor Vessel Internals

6.5.2.1 Safety Functional Requirements

The safety functional requirements of the RVI are decoupled from the safety functional requirements of the RCP [RCS].

The RVI ensures in association with other components and systems, under the specified conditions that:

a) During normal plant operation, the RVI locates, supports, restrains, protects and guides the fuel assemblies and the associated parts or components inside the RPV to support the control of reactivity function;

b) During plant normal operation and under accident conditions, the RVI protects and guides the RCCAs to ensure the control of reactivity function;

c) During plant normal operation, the RVI channels the coolant flow through the core from the RPV inlets to the RPV outlets and ensures a satisfactory distribution across the reactor core and a satisfactory core cooling to ensure the control of reactivity and heat removal functions;

d) During plant normal operation, the RVI provides the protection of the RPV and outside structures against excessive irradiation exposure from the core to support the radioactivity confinement function;

e) During plant normal operation, the RVI locates, supports, restrains, protects and guides the instrumentation within the RPV to support the extra safety function;

f) During plant normal operation, the RVI locates, supports, restrains, protects and guides the reactor vessel irradiation surveillance capsules to support extra functions;

g) Under accident conditions, the RVI provides secondary core support in case of a postulated fall of the lower internals and the core to ensure the control of reactivity function;

h) During plant normal operation, the RVI absorbs the core loads, RCCA loads and other loads, and transmits these loads to the RPV without jeopardising its integrity to support the extra functions.

6.5.2.2 Description

The reactor vessel internals consist of lower internals, upper internals and interface components. The main design parameters of the RVI are illustrated in Table T-6C-6 and the structure schematic drawing of the RVI is shown in Figure F-6D-9.

6.5.2.2.1 Lower Internals

The lower internals, which are positioned and aligned by the radial support keys and the alignment pins in the RPV, are the main supporting structures for the reactor core. The lower internals consist of the core barrel, lower support plate, metal reflector structure (i.e. core shroud) and flow distribution assembly.

The core barrel and the lower support plate are welded together to form an enclosed boundary for the reactor core. There are 4 holes corresponding to the position of each fuel assembly on the lower support plate. The lower support plate is furthermore equipped with 2 guiding pins to provide support and positioning for each fuel assembly.

The metal reflector structure is located inside the core barrel and sits on the lower support plate. It adopts an all-welded structure, which is formed by a series of { } plates.

The flow distribution assembly, which is a multi-hole hemispherical structure, is fixed on the bottom of the lower support plate by bolts. It ensures a favourable flow distribution at the reactor core inlet.

6.5.2.2.2 Upper Internals

The upper internals, providing a hold-down force for the fuel assemblies, guiding the RCCAs and the in-core instrumentation, are aligned with the lower internals by the alignment pins and alignment plates. The upper internals include the upper support assembly, upper core plate, supporting columns, Control Rod Guide Assembly (CRGA) and In-Core Instrumentation Guide Assembly (IGA).

The upper support assembly separates the upper plenum and dome plenum. It consists of the upper support plate, skirt and flange, and is welded together by circumferential welds. The upper support plate and the upper core plate form the upper plenum. There are 46 support columns between the upper support plate and the upper core plate.

68 CRGAs provide guidance for the drive rods of the CRDMs and the RCCAs.

The in-core instrumentation guide assembly is located above the upper support assembly and protects guides and supports the In-Core Instrumentation Assembly (ICIA). The 46 ICIAs are used for the In-Core Instrumentation System (RIC [ICIS]) and are divided into twelve bundles in the head plenum prior to exiting through the RPV head.

6.5.2.2.3 Interface Components

The hold down spring is a forged ring structure which is located between the upper support flange and the core barrel flange. During reactor operation, the spring maintains the vertical stability of the upper and lower RVI.

The radial support keys inserts are bolted to the rim of lower support plate, and clevis inserts are attached to the radial support keys welded to the reactor pressure vessel. They limit the rotation and tangential movement of the lower RVI. Clevis inserts and radial support keys provide a load path for the lower support plate horizontal loadings whilst allowing unrestrained radial and axial thermal growth of the core barrel and lower support plate.

6.5.2.3 Design Principles and Codes

The primary design principles for the reactor vessel internals are as follows:

a) The structures of the RVI (including the upper core plate, metal reflector structure, lower support plate, etc.) is designed to meet the safety functional requirements;

b) The selection of materials was made with mature engineering experience, and considering the following characteristics: resistance to irradiation, resistance to corrosion, resistance to abrasion and erosion and good manufacturing performance.

The structural design of the RVI complies with RCC-M and the material, NDT, welding and manufacture of RVI also conform to the requirements of RCC-M.

6.5.2.4 Classification

According to the requirements of RCC-M, component parts of the RVI are classified into two categories: Core Support Structure (CS) and Internal Structure (IS).

The CSs are those structures or parts which support and restrain the fuel assemblies to make up the core within the RPV. All other structures excluding CS are the IS.

According to the safety categorisation and classification presented in Reference [8] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of the RVI is presented in Table T-6C-4.

6.5.2.5 Materials, Manufacture and Inspection

The materials for the major components are austenitic stainless steels such as Z2 CN 19-10 (Nitrogen Controlled), Z3 CN 18-10 (Nitrogen Controlled), Z6 CND 17-12, etc., while the material for the hold down spring is Z12 CN13. The materials used for the RVI comply with RCC-M.

The filler material except for hard-facing surfaces is qualified in accordance with the requirements of RCC-M S5000. The acceptance test is performed on each lot of filler material in accordance with the requirements of RCC-M S2000.

The welding material for hard-facing surfaces is Stellite 6, which satisfies the requirements of RCC-M S8000.

For the nickel base alloy, filler metal ERNiCrFe-7/E NiCrFe-7 according to RCC-M S2981 or RCC-M S2986 is used.

Stainless steel and nickel base alloy filler material in contact with the primary coolant shall have a cobalt content of no more than 0.06%.

The CS and IS are manufactured and inspected according to RCC-M. Welds of the RVI shall meet the requirements of RCC-M. NDT of the RVI shall be performed in accordance with RCC-M and related NDT technical documents. The process qualification shall be done according to RCC-M before fabrication and surface finishing. Also, heat treatment of the RVI complies with RCC-M.

6.5.2.6 Structural Integrity

The structural integrity of the RVI is demonstrated in Reference [34] in the form of CAE from a number of aspects, which mainly include applicable design codes and standards, loading conditions, design analysis, selected material, manufacture, manufacturing inspection, operation and maintenance.

6.5.3 Control Rod Drive Mechanisms

6.5.3.1 Safety Functional Requirements

The safety functional requirements of the CRDM are as follows:

a) During plant normal operation, the CRDM can drop the RCCA into the core according to the instruction of the Rod Position Indication and Rod Control System (RGL [RPICS]) to shut down and maintain core sub-criticality to reach a controlled state or final state of plant to support the reactivity control function;

b) Under plant accident conditions, the CRDM can drop the RCCA into the core to shut down and maintain core sub-criticality to reach a controlled state or final state of plant to support the reactivity control function;

c) During plant normal operation, the CRDM contributes to the reactivity control function by inserting, withdrawing or holding the RCCA over the height of the core;

d) During plant normal operation and under accident conditions, the CRDM pressure housing assembly which constitutes a pressure boundary maintains reliable structural integrity to ensure that:

   1) The reactor coolant inventory can be maintained to support the heat removal function;

   2) The radioactive material can be confined to support the confinement function.

6.5.3.2 Description

The CRDM consists of five separate assemblies. They are the pressure housing assembly, latch assembly, drive rod assembly, coil stack assembly and rod position indicator assembly. The structure of the CRDM is shown in Figure F-6D-10 and detailed information is presented in Reference [35]. The main design parameters of the CRDM are illustrated in Table T-6C-7.

6.5.3.2.1 Structure Description

a) Pressure Housing Assembly

   The pressure housing assembly, containing the drive rod assembly and latch assembly, is composed of the latch housing and rod travel housing. It is a part of the pressure boundary and plays an important role in containing the radioactive products. It provides mechanical support for the latch assembly and moving space for the drive rod assembly. The pressure housing assembly is installed on the CRDM adapter of the RPV head and is connected by a threaded, seal-welded, maintainable joint that facilitates disassembly.

b) Latch Assembly

The latch assembly is assembled inside the pressure housing assembly. It includes the guide tube, lift pole, stationary latch pole, movable latch pole, two sets of latches, etc. The latch adopts a double-tooth structure to meet the requirement of cumulative stepping number. Each latch assembly has 6 latches, which are divided into two sub-assemblies, a stationary latch sub-assembly and a movable latch sub-assembly. In each sub-assembly, 3 latches are uniformly distributed on the circumference (each latch is 120 degrees apart from the others). The latches engage with the grooves of the drive rod assembly. The movable latch sub-assembly provides vertical movement between a high position and a low position to move the drive rod and RCCA to the required position in stepped increments. Once the coils are de-energised, the latches open rapidly under the force of the springs and gravity, so that the RCCA could be released into the reactor core to achieve the function of reactor fast shutdown.

c) Drive Rod Assembly

The drive rod assembly includes the coupling, drive rod, disconnect button, disconnect rod, locking button, etc. The drive rod is designed with circumferential grooves to engage with the latches during the holding and moving of the drive rod.

The coupling is attached to the drive rod and provides the means of connection to the RCCA directly below the CRDM. The disconnect button, disconnect rod, and locking button provide positive locking of the coupling to the RCCA and permit remote disconnection of the RCCA.

d) Coil Stack Assembly

The coil stack assembly is placed outside the latch housing and includes the coil housings, electrical conduit, connector, three operating coils, etc. The operating coils are connected to the RGL [RPICS] via an electrical connector. Energising the operating coils causes the movement of the latch assembly.

The maximum design operating temperature of the coils is 200°C. So the coil stack assembly is cooled by forced air cooling when in operation to make sure that the operating coils are below their design operating temperature of 200°C. A loss of the cooling air would result in the release of the drive rod in the worst case scenario.

e) Rod Position Indicator Assembly

The rod position indicator assembly is located outside the rod travel housing. It consists of the primary coil, secondary coil, auxiliary coil, protective sleeve, connecting flanges, electrical connector, top clamping ring, lower end, etc.

The rod position indicator assembly is used to indicate the actual position of RCCA inside the core. During the rod drop test, the rod position indicator is also used to measure drop time. When the drive rod is moved upwards and downwards inside the rod position indicator coils, the actual position of the drive rod top end can be indicated by measuring the output voltage of the rod position indicator coils.

The rod position indicator assembly is connected to the RGL [RPICS]. The actual position of the RCCA can be provided to the operator by the processed signal generated by the RGL [RPICS].

6.5.3.2.2 Functional Description

The sequences presented in Tables T-6C-8 and T-6C-9 describe the withdrawing and inserting of the RCCA in one step from the hold position in which only the stationary coil is energised.

In case of an interruption in the power supply to the coils, the armatures fall down, the latches open and the drive rod with the RCCA falls into the core due to gravitational force.

The interfaces of the CRDM include the following items:

a) Coupling with the RCCA;

b) Thread and Canopy weld connection with the RPV head adapter;

c) Interfaces with the CRGA;

d) Interfaces with the reactor head package, the including seismic bearing device and ventilation hood;

e) Interfaces with the RGL [RPICS];

f) Interfaces with the containment cooling and ventilation system.

6.5.3.3 Design Principles and Codes

6.5.3.3.1 Mechanical Design Requirements

The sealing device of pressure parts is designed to be safe and reliable enough to assure the integrity of the pressure boundary. It also needs to be designed to allow for disassembly.

The structure is designed to be convenient for disassembly, overhaul and the replacement of inner parts.

The drive rod assembly is designed to reliably connect to the RCCA. The connection and disconnection is able to be performed expediently by remote operation.

It needs to be guaranteed that the RCCA can be released under gravity without any further operations when the coils are de-energised, so that the emergency shutdown function can be achieved.

The design life of moving parts in the CRDM can meet the requirement of the cumulative stepping number.

The structural design of the CRDM complies with RCC-M and the material, NDT, welding and manufacture of the CRDM also conform to the requirements of RCC-M. PSIs and ISIs of CRDM components follow the stipulations of the RSE-M rule.

6.5.3.3.2 Electrical Component Design Requirements

The electrical components are designed to be independent modules which are convenient for checking and replacing.

The metallic material, insulation material and sealing material adopted are able to work normally in high temperature and a high radiation environment. The performance is verified by identification tests or operating experience in the in-service NPP.

The electrical component is able to provide good waterproof properties in the normal working conditions.

6.5.3.4 Classification

According to the method of safety categorisation and classification presented in Reference [8], the classification of the CRDM is presented in Table T-6C-4.

6.5.3.5 Materials, Manufacture and Inspection

Metallic materials used in the CRDM are austenitic stainless steel, martensitic stainless steel, nickel-based alloy, cobalt-based alloys, ductile iron and carbon steel. The pressure boundary components are made of the material Z2 CN 19-10 controlled nitrogen content austenitic stainless steel. The non-pressure boundary components in contact with the primary coolant are made of the materials of Z2 CN 19-10 controlled nitrogen content austenitic stainless steel and Z5 CN 18-10 austenitic stainless steels, X12 Cr 13 and X12 CrNi 13 martensitic stainless steels, as well as nickel-based alloys and cobalt-based alloys. The non-pressure boundary components in contact with cooling air are made of XC10, ductile iron, etc. The controlled nitrogen content austenitic stainless steel used in the pressure housing assembly complies with RCC-M Subsection M. Materials used in magnetic fields are selected with proper magnetic performance.

The material for hard-facing surfaces is Stellite 6, the acceptance tests are performed in accordance with the requirements of RCC-M S8000.

For the canopy weld, filler material ER308L is used and the acceptance tests are performed on each lot of filler material in accordance with the requirements of RCC-M S2910.

Fabrication of pressure components is performed in compliance with RCC-M Subsection B.

The non-destructive inspection tests of the CRDM during fabrication comply with the requirements of RCC-M Subsection MC.

6.5.3.6 Structural Integrity

As part of the reactor coolant pressure boundary, the CRDM pressure housing assembly is designed to ensure its structural integrity. The pressure housing assembly belongs to RCC-M class 1. As a class 1 part, the design of the pressure housing assembly is governed by the requirements of RCC-M Subsection B. The pressure housing assembly and its welds are designed, analysed, fabricated, inspected, and tested in accordance with the requirements of RCC-M.

The structural analysis, seismic analysis, and fatigue analysis of the pressure housing assembly are performed according to the requirements of RCC-M.

In the factory of the supplier, the pressure housing assembly is subjected to hydraulic tests. The test pressure and test method comply with the requirements of RCC-M B5000.

6.5.3.7 Qualification

The operational function and lifetime of the CRDM is qualified in the CRDM qualification test, including the life test and seismic test. These tests are performed on a test bench of the control rod drive line. The test program also includes a rod drop test. The test program is chiefly designed to demonstrate correct reactor control and shutdown rod behaviour. Relevant qualification schedule is detailed in Reference [24].

6.5.3.8 Examination and Maintenance

In order to establish the condition of the CRDM during the plant normal operation stage, a series of tests can be performed during outage. This refers to electrical tests, such as a coil resistance test and an insulation resistance test.

The PSIs and ISIs of the canopy weld are subjected to the requirements of the RSE-M code. The rod drop test is performed for all of the CRDMs to check the drop time after refuelling.

If necessary, the CRDM can be disassembled and replaced entirely or partly.

6.5.4 Steam Generator

6.5.4.1 Safety functional requirements

The steam generator is designed to fulfil the following safety functional requirements:

a) Ensuring the pressure boundary integrity during normal, upset, emergency and fault conditions through its design life;

b) Serving as the first means for removal of heat from the reactor. The steam generators transfer the core thermal power (entirely or a fraction), the Reactor Coolant Pump power and the stored heat in the fluid and the metallic structures to the main secondary system;

c) Providing the reactor with a minimum heat sink reserve (SG water inventory) to ensure the mitigation of all plant conditions/transients with respect to their associated criteria.

d) Acting as an expansion vessel for the secondary water supply systems, so as to protect the secondary steam equipment against liquid flow (including the turbine at power) after the appropriate isolation of the overfeeding source, for example in case of a SGTR event.

e) Producing steam with no more than 0.1% moisture carry-over at the steam generator outlet (before the steam flow limiter) to prevent damage to the turbines using reactor coolant as the heat source.

f) Providing access to the measuring instrumentation such as water level indications and automatic control of water level at any power level and during all operation conditions.

6.5.4.2 Description

The steam generator is a natural circulation U-tube heat exchanger with separation equipment in the steam drum region. The main design parameters of the SG are shown in Table T-6C-10.

The steam generator is comprised of two subassemblies: the lower subassembly ensures vaporisation of the feedwater and the upper subassembly ensures the mechanical drying of the steam-water mixture produced in the lower subassembly. The steam generator is arranged vertically and the steam water mixture flows upward by natural circulation. The feedwater enters the steam generator through the main feedwater nozzle, which is positioned in the lower region of the steam drum. The feedwater internal header arrangement is designed to reduce the potential for thermal stratification of feedwater within the thermal sleeve and header. Feedwater is directed through the distribution ring header which is equipped with J-tubes on the top of the header to prevent draining and to direct feedwater downwards into the downcomer.

The water flows down and enters the tube bundle region, then turns and flows upward along the tube bundle. As it flows upward, the fluid is heated and becomes a mixture of saturated liquid and steam. This saturated mixture continues to flow upward into the moisture separator assemblies, the liquid flow is returned to the SG downcomer and the steam leaves the SG through the steam outlet nozzle.

The steam generator is designed to access the tube bundle for inspection and maintenance.

6.5.4.2.1 Lower Subassembly

The lower subassembly consists of:

a) The channel head

The channel head is mainly constituted of a hemispherical bottom head. A short cylindrical section is included at the channel head upper part (between head and tubesheet) to improve access to peripheral tubes for inspection. A partition plate divides the channel head into two leak tight chambers. One chamber is connected to the reactor vessel outlet (steam generator inlet or hot leg) and the other to the reactor vessel inlet (steam generator outlet or cold leg) via the Reactor Coolant Pump (which is positioned on the cold leg between the SG and the RPV). Each chamber includes a nozzle and a safe end, for connection to the reactor coolant system, and a manway, which allows access for in-service inspection and maintenance.

b) Tubesheet

Tubesheet is an integrated forging piece and the primary side of the tubesheet is cladding with Inconel and stainless steel materials. The tubesheet holes are triangular pitch type.

c) The secondary shell

The secondary shell is formed of two cylindrical shells and a conical shell. The shell is fitted with 4 hand holes in the lower shell just above the tubesheet for in-service inspection and maintenance in the lower part of the tube bundle. The shell is also equipped with two diametrically opposite hand holes at the top (9th) tube support plate, on the axis of the tube lane, to permit U-bend region inspections and maintenance operations, a set of nozzles for water level measurement and two blowdown nozzles within the tubesheet.

d) The tube bundle

The tube bundle of the inverted U type tubes enables heat exchange between the reactor coolant, circulating inside the tubes, and the secondary fluid. It also functions as a radiological barrier between the primary and secondary sides of the Nuclear Steam Supply System (NSSS). The tube bundle arrangement is of the triangular pitch type. The ends of the inverted U-tubes are welded to the tubesheet cladding. These welds are submitted to a helium leak test and the ends of the U-tubes are then full depth expanded into the tubesheet to minimise gaps. The tube expansion process is demonstrated to minimise residual stresses in the transition from the expanded zone to the unexpanded zone. Measures are taken to assure that tubes are expanded close to but just below the secondary side of the tubesheet.

e) The lower internals

The lower internals support the tube bundle while ensuring the secondary fluid flow circulation. The lower internals include a bundle shroud (wrapper), tube supports properly spaced over the straight leg of the tube bundle and the U-bend tube supports which consist of a series of anti-vibration bar assemblies positioned between each layer of tubes.

6.5.4.2.2 Upper Subassembly

The upper subassembly (steam drum) is formed of a cylindrical shell and elliptical upper head.

The cylindrical shell includes:

a) Two manways which allow access to the steam/moisture separation assembly. These also allow access to the feedwater gooseneck and to the top of the tube bundle through a hatch in the primary deck supporting the primary steam/water separators;

b) A set of water level and steam pressure measurements taps.

The elliptical upper head includes an integrally forged (without any weld) steam outlet nozzle fitted with a welded steam flow limiter which limits the forces on the SG bundle and internals in case of a steam line break event.

The steam drum region is equipped with:

a) Steam/water separation assembly including:

   1) A first stage of cyclone type separators (connected to the primary deck which forms the top of the riser/bundle wrapper region), creating high quality steam and minimizing carry-under of steam into the recirculated water;

   2) A secondary stage of cyclone type separators/dryer units which further reduce the carry-over content of moisture at SG outlet.

b) A main feedwater assembly including:

   1) The main feedwater nozzle in the lower portion of the drum cylindrical shell to reduce the chances that the header is uncovered during low water levels;

   2) The thermal sleeve and gooseneck piping, which help to prevent thermal stratification and water hammer;

   3) An integral loose parts trapping system, incorporated into the J-tubes, which prevent intrusion of loose parts being carried along by the feedwater into the SG secondary side.

c) An emergency feedwater assembly including:

   1) Its specific inlet nozzle;

   2) The thermal sleeve;

   3) The emergency feedwater injection pipe with a similar gooseneck and downturned outlet end is to prevent any risk of impact of cold water onto the

pressure retaining shells.

6.5.4.2.3 Support

The steam generator support system is designed to allow thermal expansion of the loop and pressure displacement, but limits such displacement during accidents. The steam generator is supported vertically by four support legs and laterally at two levels, one at the tubesheet and one just below the conical shell. Figure F-6D-11 is the schematic diagram of the steam generator support system.

6.5.4.2.4 SG Insulation

SG is insulated with the cassette-type insulation filled with stainless steel foils (RMI type).

6.5.4.3 Design Principles and Codes

The steam generator provides high quality steam for the NSSS, to guarantee the stable and reliable operation of the plant. The design principles of the steam generator are as follows:

a) The SG design meets the safe functional requirements described in Sub-chapter 6.5.4.1;

b) The SG design is able to prevent unacceptable U-tubes damage caused by mechanical or flow induced vibration;

c) The selection of materials based on mature engineering experience, and considering the compatibility with the surrounding environment, such as chemical corrosion, stress corrosion and other conditions;

d) The ISI requirements are established to ensure that the effective inspection and maintenance of the steam generator during operation.

For the steam generator, the ASME Code is selected as the code of design, fabrication, inspection and testing. Pre-service and in-service inspection of the SG is performed according to RSE-M. The SG safe ends connected to the main coolant lines are of a material compliant with RCC-M code.

6.5.4.4 Classification

According to the safety categorisation and classification presented in Reference [8] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of the SG is presented in T-6C-4.

6.5.4.5 Thermo-hydraulic design

6.5.4.5.1 Thermodynamic Criteria

The steam generator is designed so that excessive oscillations of the water level do not occur during operation.

The circulation ratio is defined as the ratio of total riser flow rate within the tube bundle to steam flow rate output of the steam generator. The nominal circulation ratio offers a good compromise between efforts to maintain high enough flow to give stable operation and prevent excessive deposit build up or localized dry out, while not being so high as to cause deleterious flow induced vibration of the tubes. A higher circulation ratio also favours the steam generator transient behaviour by minimising water level shrink in case of transients (turbine or reactor trip).

The steam/water separators are very accommodating of fluctuations in the steam/water and water level throughout loading. However, moisture carryover will eventually reach a "break point" due to either a very high water level (flooding of primary separator) or a very high water loading, at which point the carryover increases rapidly. The separators are designed so that this break point occurs at a load/level well beyond the steam generator nominal operating point. The moisture mass content of the steam is under 0.1% at the outlet nozzle (before the flow limiter) of the steam generator under normal operation.

6.5.4.5.2 Thermal Design

Extensive analysis supports the design insofar as its ability to produce the required steam mass flowrate and pressure at full power for the specified conditions of reactor coolant flow and temperature. Both one-dimensional (1-D) and three dimensional (3-D) analyses are performed to understand, model and quantify the performance of the steam generator. The basic thermal-hydraulic sizing of the steam generator is performed using classical analysis techniques and one-dimensional thermal-hydraulic code analysis. Three-dimensional analysis results are utilised in specific areas that require detailed knowledge of the various flow parameters (e.g. flow induced vibration, understanding steam quality at various regions throughout the bundle, especially the U-bend area and velocities at the top of tubesheet as they relate to sludge deposition). In addition, the heat transfer results from the 3-D thermal-hydraulic analysis provide additional confidence in the results obtained from the 1-D analysis

6.5.4.5.3 Hydraulic Design

The hydraulic design is conducted in order to obtain:

a) Stable circulation;

b) An acceptable flow distribution in the steam generator secondary side, notably in relation to the risks of sludge deposition, corrosion-erosion as well as tube bundle vibrations;

c) Sufficient secondary water inventory to meet heat sink requirements.

Absence of water level oscillations at the steam-water interface is inherent in the design as confirmed by good operating experience of many similar designs. Notably, there is an adequate stability ratio ({ }) between single phase losses and two phase losses in the recirculation loop.

To limit accumulation of sludge on the secondary side and associated induced risks of stress corrosion cracking for the tube bundle, particular attention is paid to reduce the areas of low flow velocities, notably above the tubesheet and within the tube supports. This is achieved by understanding the hydraulics within all regions of the steam generator, and specifically achieving a relatively high (but not too high) circulation ratio. This results in good sweeping flow across the tubesheet to minimise sludge deposition at the secondary face, and continuing healthy flow up through the bundle. Also, a sludge collector is incorporated into the primary deck. This device will capture and sequester sludge in to a more benign region of the steam generator away from the tube bundle.

The potential for flow assisted corrosion-erosion degradation is avoided by the proper selection of materials.

6.5.4.5.4 Tube Bundle Vibration

In the design of steam generators, the possibility of degradation of tubes due to either mechanical or flow induced excitation is thoroughly evaluated. This evaluation includes detailed analysis of the tube support system supported by an extensive research program with tube vibration model tests.

In evaluating the risk of failure due to vibration, consideration is primarily given to the source of excitation coming from secondary fluid flow on the outside of the tubes. During normal operation, the effects of primary fluid flow within the tubes and mechanically induced vibration are negligible. In general, three vibration mechanisms have been identified:

a) Vortex shedding resonance

Vortex shedding, when the frequency of the wake of shedding vortices matches (or is close to, i.e. within about { }) the natural frequency of a span of tube, is difficult to avoid completely, especially in the bundle entrance region where single phase fluid is primarily in cross flow against the outer few rows of tubes in the bundle. Notwithstanding, the following considerations help to deter the mechanism:

1) Flow turbulence in the down-comer and tube bundle inlet region inhibits the formation of Von Karman's vortex train;

2) The spatial variations of cross flow velocities along the tube preclude vortex shedding at a single frequency;

3) Both axial and cross flow velocity components exist on the tubes. The axial flow components disrupt the Von Karman vortices.

b) Fluid elastic instability

Concerning fluid elastic instability, when the tube cannot absorb the energy being applied, the stability ratio is not greater than 0.75 assuming all supports are effective.

c) Random turbulence excitation

Turbulence in single and two-phase flow causes continuous, small amplitude, broadband vibration. This effect is important and is evaluated for the at-risk regions of the bundle entrance and U-bend. Wear analysis is done in conjunction with this assessment to show acceptable tube wear over the design life of the SG.

The above described mechanisms are kept to acceptable levels, or prevented, through both careful hydraulic design and mechanical design.

Hydraulically, the circulation ratio is balanced to achieve a high enough value to minimise deposition at the top of the tubesheet and at tube support contact locations, and prevent dry-out at susceptible locations, while not creating too high a flow to produce vibration problems.

Mechanically, tubes are supported by the tube supports. Design clearances, achieved with careful control of the manufacturing assembly operations, are set to be equal to or better (i.e. tighter) than industry established values. Good contact length ({ }) with tube-touching points being smooth and round, and proper material selection (quenched and tempered Type 410S stainless steel), ensures minimal potential for fretting.

6.5.4.6 Materials, manufacture and Inspection

6.5.4.6.1 Materials

All pressure boundary materials used in the steam generator are selected and fabricated in accordance with the requirements of the applicable codes. High strength low alloy ferritic steel forgings (SA-508 Grade 3 Class 2 or other equivalent material) are selected for the main pressure boundary parts, including the primary head, tubesheet, cylindrical shells, conical shell, upper elliptical head and major nozzles. Alloy 690 is selected for the tubes of the tube bundle. The tubing procurement specification is mature and incorporates best practices for achieving a very high quality tube material. The primary head divider plate is made of Alloy 690. The tube support plates and U-bend supports are made of corrosion-resistant 13% Cr martensitic stainless steel of Type 410S or other equivalent material.

For pressure retaining welds, and nickel and stainless steel overlay on the tubesheet and primary head, the weld consumables meet the requirements specified in ASME code Section II Part C as a minimum. The filler materials used for welding SA-508 Grade 3 Class 2 are SFA-5.23 F9P4-EG-G and/or SFA-5.5 E9018-B3, which have equivalent mechanical properties. The surface of the tubesheet is cladded with ERNiCrFe-7 and ENiCrFe-7 for the primary side and 308L/309L for the straight edge region. The surface of the primary head is cladded with Type 308L/309L austenitic stainless steel to avoid contact with the primary coolant. ERNiCrFe-7 and/or ENiCrFe-7, which are highly resistant to Primary Water Stress Corrosion Cracking (PWSCC), are selected for welding the welds including the buttering between the primary inlet/outlet nozzle and their safe ends. No filler material is used for tube-to-tubesheet welds, except that ERNiCrFe-7 is used for repairing these welds.

6.5.4.6.2 Manufacturing

The main pressure boundary parts are forgings. The material surfaces in contact with the primary coolant are made of or cladded with austenitic stainless steel or Ni-Cr-Fe Alloy. The ends of the inverted U-tubes are welded to the tube-sheet cladding. The U-tubes are expanded inside the full depth of the tubesheet by using a hydraulic process. A helium leak check is conducted for the seal weld of each tube before the full depth expansion. In the manufacturing stage, the primary side and secondary side of the steam generator surface is clean.

6.5.4.6.3 Inspection

The material is examined in accordance with the material procurement specifications. As a minimum, the examinations and tests required by ASME Section III, Division 1, Subsection NB-2000 are included, along with any additional buyer specific requirements.

While in fabrication, the steam generators are examined and tested in accordance with ASME Section III, Subsection NB-5000, and any additional customer specified requirements.

6.5.4.7 Structural Integrity

The structural integrity of the SG is demonstrated in Reference [36] in the form of CAE from a number of aspects, which mainly include applicable design codes and standards, loading conditions, design analysis, selected material, manufacture, manufacturing inspection, operation, maintenance and the defect tolerance assessment.

6.5.5 Pressuriser

6.5.5.1 Safety Functional Requirements

The main safety functional requirements of the PZR are as follows:

a) During plant normal operation, the PZR homogenises the boric concentration between primary loop and PZR through spraying and heating operations, to support the reactivity control function;

b) During plant normal operation, the PZR provides adequate volume to ensure that the water volume change due to reactor coolant temperature fluctuation can be compensated, to support the heat removal function;

c) During plant normal operation and under accident conditions, the PZR (spraying and heating) performs a pressure control function in order to maintain the over-cooling margin of the reactor coolant, to ensure the heat removal function;

d) During plant normal operation and under accident conditions, the PZR provides reliable integrity and serves as part of the pressure boundary, to perform confinement function;

e) Under accident conditions, the PZR performs a overpressure protection function of the primary loop via PSVs, to support confinement function;

f) Moreover, during plant normal operation and under accident conditions, the PZR performs monitoring functions for the reactor coolant inventory (reflected by water level in PZR) as well as the RCP [RCS] pressure (reflected by PZR pressure), to ensure the extra supporting functions.

6.5.5.2 Description

As shown in Figure F-6D-12, the PZR is a vertical vessel. The bottom of the PZR is connected to the SL. The main parameters are presented in Table T-6C-11. The PZR consists of a cylinder assembly, upper and lower head assemblies. The upper head assembly, cylinder assembly, and the lower head assembly are welded together.

a) Upper Head Assembly

The upper hemispherical head is a forged single-piece. It is equipped with one venting nozzle, three safety valve nozzles which connect to the pressure safety valves and take part of the overpressure protection function, one severe accident safety valve nozzle connected to the severe accident dedicated valve and one spray nozzle welded on the top of the upper hemispherical head which takes part of the pressure control function.

b) Cylinder Assembly

The cylinder assembly consists of three forged sub-components (upper cylinder shell, middle cylinder shell and lower cylinder shell), measurement taps which help to measure the water level, pressure and temperature of the PZR and a manway which allows access to the PZR interior for inspection and maintenance.

c) Lower Head Assembly

The lower hemispherical head is a forged single-piece. It is equipped with one surge nozzle at the bottom of the PZR which connects to the SL, measurement taps which help to measure the water level and the temperature of the PZR and 108 heaters which take part of the pressure control function.

d) Support System

The PZR vessel is supported by 3 lower vertical supports which are welded on the lower cylinder shell. In addition, 8 horizontal limiters are also provided on the upper cylinder shell to maintain the integrity of the PZR under all bounding load conditions. The structure schematic drawing of the PZR support system is presented in Figure F-6D-13.

e) PZR Insulation

The PZR is insulated with the cassette-type insulation filled with stainless steel foils (RMI type).

6.5.5.3 Design Principles and Codes

The PZR is designed according to internationally recognized codes and standards. Operating experience is also taken into account. The PZR design is similar to typical pressurised water reactor PZR designs. Similar proven materials and manufacturing processes are used.

The design of the PZR considers the following basic principles:

a) The structure of the PZR complies with the safety functional requirements;

b) Selecting materials with mature international engineering experience, and considering the compatibility with the surrounding environment, such as chemical corrosion and other conditions;

c) Using large forgings as far as possible to reduce the number of welds in the reactor coolant pressure boundary;

d) Facilitating manufacturing, inspection and maintenance.

The structural design of the PZR complies with the provisions of RCC-M Subsection B. The material, NDT, welding and manufacture of the PZR respectively comply with the requirements of RCC-M Subsection M, Subsection MC, Subsection S and Subsection F. Pre-Service Inspection and In-Service Inspection of the PZR are in accordance with the stipulations of RSE-M.

6.5.5.4 Classification

According to the safety categorisation and classification presented in Reference [8] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of the PZR is presented in Table T-6C-4.

6.5.5.5 Materials, Manufacture and Inspection

The main material of the PZR is 18MND5, and the inner surface which is in contact with reactor coolant is cladded with the austenitic stainless steel. The material of the nozzle safe end, measurement tap, heater sleeve, connecting part of the heaters and the end plug of heaters is Z2CND18.12 (Nitrogen Controlled). The material of the heater sheath, thermal sleeve and heater support plate is Z2CND17.12. The selected materials are fully considered using engineering application experience. The procurement and manufacture of the materials comply with the requirements of RCC-M. The chemical element contents, such as Copper, Sulphur, Vanadium (for shells), Phosphorus (for nozzles) and Cobalt (for the cladding), are more strictly controlled than in the RCC-M requirements.

The filler material is qualified in accordance with the requirements of RCC-M S5000. The acceptance test is performed on each lot of the filler materials in accordance with the requirements of RCC-M S2000.

For low alloy steel welds, filler material according to RCC-M S2820 and S2830 are used.

For dissimilar metal welds or buttering, filler metal alloy ERNiCrFe-7/E NiCrFe-7 according to RCC-M S2981 or RCC-M S2986 is used.

For corrosion resistant cladding, filler material 309L according to RCC-M S2930 or RCC-M S2970 is used and 308L according to RCC-M S2920 or RCC-M S2960 is used.

Stainless steel and nickel base alloy filler material in contact with the primary coolant are to have a cobalt content of no more than 0.06%.

The forgings of the PZR components are manufactured according to the procedures qualified under the requirements of RCC-M Subsection M. The welding process is qualified according to RCC-M Subsection S. The machining of components complies with the requirements of RCC-M Subsection F.

The NDTs of the PZR components during the fabrication comply with the requirements of RCC-M Subsection MC.

The Pre-Service Inspection (PSI) and In-Service Inspection (ISI) are subject to the requirements of RSE-M code involving welds and positions to be inspected, range of such positions and required methods for inspection. The non-destructive inspection equipment, procedure and personnel are qualified and ensure defect detectability under actual inspection conditions to guarantee the effectiveness and reliability of non-destructive inspection techniques and accurate results.

6.5.5.6 Structural Integrity

The structural integrity of the PZR is demonstrated in Reference [37] in the form of CAE from a number of aspects, which mainly include applicable design codes and standards, loading conditions, design analysis, selected material, manufacture, manufacturing inspections, operation, maintenance and defect tolerance assessment.

6.5.6 Reactor Coolant Piping

6.5.6.1 Safety Functional Requirements

The main safety functional requirements of the reactor coolant piping are as below:

a) Reactor coolant piping does not perform the reactivity control function directly. However, during plant normal operation and under accident conditions, it serves as routes to ensure the borated water can be injected into or adjusted in the reactor core by the auxiliary systems or safety systems, to support the reactivity control function;

b) During plant normal operation and under accident conditions, the reactor coolant piping serves as routes to ensure reactor coolant can be conveyed between the RPV, SGs and Reactor Coolant Pumps or interfacing systems of the RCP [RCS], to support the heat removal function;

c) During plant normal operation and under accident conditions, the reactor coolant piping provides reliable integrity and serves as part of the reactor coolant pressure boundary, to perform the confinement function;

d) During plant normal operation and under accident conditions, the reactor coolant piping provides a monitoring interface to ensure the important system operating parameters can be monitored (e.g. reactor coolant temperature, flow rate of Reactor Coolant Pumps, and primary loop water level when the RCP [RCS] are drained for maintenance or refuelling), to support the extra supporting functions.

6.5.6.2 Description

Reactor coolant piping belongs to the NSSS. It consists of two parts: the MCLs and the SL. The main parameters for both the MCLs and SL are shown in Table T-6C-12.

a) Main Coolant Lines

As shown in Figure F-6D-14, the MCLs carry reactor coolant from the RPV to the SG and then to the Reactor Coolant Pump. The fluid is then finally returned to the RPV. There are three reactor coolant loops, each comprising:

1) One Hot Leg (HL): connects the RPV to a SG. It is made of one forged piece which comprises of straight sections and one elbow;

2) One Crossover Leg (UL): connects a SG to a Reactor Coolant Pump. It is made of three forged pieces, which comprises of straight sections and three elbows;

3) One Cold Leg (CL): connects a Reactor Coolant Pump to the RPV. It is made of one forged piece which comprises of straight sections and one elbow.

b) Surge Line

As shown in Figure F-6D-15, the SL (which consists of 6 parts) connects the hot leg of loop 3 to the PZR. It is designed to alleviate thermal stratification under steady-state operation. The supporting system for the SL consists of two support hangers at two fixed positions on the SL. One end of the support hanger is fixed on the SL and the other end is composed of a spring supporting frame welded directly or through a bracket on the embedded plate.

c) Insulation

The MCLs (CL, HL and UL) and SL are insulated with the cassette-type insulation filled with stainless steel foils (RMI type).

6.5.6.3 Design Principles and Codes

The reactor coolant piping is designed according to internationally recognised codes, standards and takes into account operating experience. The reactor coolant piping design is similar to typical pressurised water reactor relevant designs, which uses similar proven materials and manufacturing processes.

The design of the reactor coolant piping considers the following basic principles:

a) The structure of reactor coolant piping complies with the safety functional requirements;

b) Selection of materials according to mature international engineering experience, and considering the compatibility with the surrounding environment, such as chemical corrosion and other conditions;

c) Reducing the welds in the reactor coolant pressure boundary as far as possible;

d) The design of the SL eliminates or alleviates thermal stratification as far as possible;

e) Ensuring that the design facilitates manufacturing, inspection and maintenance.

The structural design of the reactor coolant piping complies with the provisions of RCC-M Subsection B. The material, NDT, welding and manufacture of the reactor coolant piping respectively comply with the requirements of RCC-M Subsection M, Subsection MC, Subsection S and Subsection F. The PSI and ISI of reactor coolant piping are in accordance with the stipulations of RSE-M.

6.5.6.4 Classification

According to the safety categorisation and classification presented in Reference [8] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of reactor coolant piping is presented in T-6C-4.

6.5.6.5 Materials, Manufacture and Inspection

The material of the MCLs forging pipe (including the integrated nozzle) is X2CrNi19.10 (Controlled Nitrogen Content) in accordance with RCC-M M3321, and the material of welded nozzles is Z2CN19.10 (Controlled Nitrogen Content) in accordance with RCC-M M3301. The material of the SL is X2CrNiMo18.12 (Controlled Nitrogen Content) in accordance with RCC-M M3321. The selected materials consider the engineering application experience. The procurement and manufacturing of the material comply with the requirements of RCC-M. The chemical element content of Cobalt is more strictly controlled than in the RCC-M requirements.

The filler material is qualified in accordance with the requirements of RCC-M S5000. The acceptance test is performed on each lot of filler material in accordance with the requirements of RCC-M S2000.

Filler material E316L and ER316L according to RCC-M S2925 or RCC-M S2915 is used for welding of MCLs and SL. In addition, the cobalt content of weld metal is no more than 0.06%.

The forgings of the reactor coolant piping components are manufactured according to the certificated processing procedures qualified according to the requirements of RCC-M Subsection M. The welding process is qualified according to RCC-M Subsection S. The machining of components complies with the requirements of RCC-M Subsection F.

During the fabrication, the NDTs of the reactor coolant piping components comply with the requirements of RCC-M Subsection MC.

The PSI and ISI are subject to the requirements of RSE-M code involving welds and positions to be inspected, range of such positions and required methods for inspection. The non-destructive inspection equipment, procedure and personnel shall be qualified and ensure defect detectability under actual inspection conditions to guarantee the effectiveness and reliability of non-destructive inspection techniques and accurate results.

6.5.6.6 Structural Integrity

The structural integrity of MCL is demonstrated in Reference [38] in the form of CAE from a number of aspects, which mainly include applicable design codes and standards, loading conditions, design analysis, selected material, manufacture, manufacturing inspections, operation and maintenance.

6.5.7 Reactor Coolant Pumps

6.5.7.1 Safety Functional Requirements

The safety functional requirements of the Reactor Coolant Pumps are as outlined below:

a) During plant normal operation, the Reactor Coolant Pumps shall provide an adequate enforced flowrate to ensure the heat removal function;

b) Under plant accident conditions, the Reactor Coolant Pumps shall provide an adequate enforced flowrate to support the heat removal function;

c) Under plant accident conditions combined with a LOOP, the Reactor Coolant Pumps shall provide an adequate inertia flow rate at the early stage after reactor trip to support the heat removal function;

d) The assemblies which constitute the pressure retaining boundary shall provide reliable radioactive confinement functions, which mainly includes the:

1) Pump casing;
2) Shaft seal assembly.

6.5.7.2 Description

The Reactor Coolant Pump is comprised of a vertical, single stage pump and an asynchronous motor which is located above the pump. The main design and performance parameters are shown in Table T-6C-13, and the structure schematic diagram is shown in Figure F-6D-16. The Reactor Coolant Pump will be detailed designed and constructed by the equipment vender according to the Reactor Coolant Pump equipment specification which defines the requirements of design, material, fabrication, inspection and examination, test, etc.

The three major parts of the Reactor Coolant Pump are the hydraulic unit, shaft seal assembly and the motor:

a) The hydraulic unit is made up of a casing, an impeller, a diffuser and a suction adapter. The other elements which support the hydraulic unit are the shaft, the guide bearing, the high pressure cooler, the coupling, the spool piece and the motor stand;
b) The shaft seal assembly is made up of three stages of identical hydrodynamic mechanical seals in series and one stage of standstill seal;
c) The motor is a squirrel cage induction motor, protected from water spray, which is made up of a solid shaft, a double thrust bearing, an upper and a lower oil film radial guide bearings, an anti-rotation device and a flywheel.

The target service life of the Reactor Coolant Pump is 60 years, excluding the wearing parts, such as the shaft seal assembly, bearings, gaskets and O-rings, etc. According to the design and operating experience of the Reactor Coolant Pumps, the parts list of the Reactor Coolant Pump along with the service life is given in Table T-6C-14.

All parts in the Reactor Coolant Pump can be replaced for maintenance except the casing which is welded to the Reactor Coolant Pipes. The internal components of the Reactor Coolant Pump can be easily removed from the casing.

6.5.7.2.1 Hydraulic Unit

The pump suction is in the axial direction and the pump discharge is in radial direction. The reactor coolant enters the suction nozzle of the casing, is directed towards the impeller by the suction adapter, passes through the impeller and exits through the diffuser and the discharge nozzle.

The casing is made of low alloy steel forging with a stainless steel cladding layer. There is an axial suction nozzle welded to the Cross-over Leg and a radial discharge nozzle welded to the Cold Leg of the casing. According to Reference [39], the use of forged low alloy steel pump casings with welded support lugs is appropriate for the UK HPR1000. The casting casing is also a possible option according to the experience of the pump supplier. If the future operator prefers the casting casing, relevant safety analysis and demonstration of casting casing should be review in site licensing stage.

The impeller is an integral part made of stainless steel and connected to the pump shaft. The impeller is the main component of the Reactor Coolant Pump's hydraulic parts. It transfers the mechanical energy to hydraulic energy through rotation. The diffuser is made of stainless forging, and is between the casing and the impeller. With the diffuser the hydrodynamic head of the pump is transferred to the hydrostatic head.

The shaft and the internal static parts are provided with a seat seal to prevent leakage from the RCP [RCS] when the motor shaft is uncoupled and seated.

A guide bearing is installed between the shaft seal assembly and the impeller. It is a hydrodynamic swivel bearing. In normal operation, the bearing is lubricated and cooled by the seal injection water.

The spool piece is located between the pump shaft and the motor shaft, so that the maintenance of the shaft seal assembly can be done without removing the motor.

The studs and nuts for the casing and seal housing are designed to be tightened by hydraulic tensioning or by heating.

6.5.7.2.2 Shaft Seal Assembly

The shaft seal assembly resists the entire pressure drop of primary loop of the RCP [RCS] and controls the leakage between the rotating parts and stationary parts of the Reactor Coolant Pump. The shaft seal assembly consists of three stages of hydrodynamic mechanical seals in series, and one standstill seal downstream from the third stage of the seal.

Each stage of the hydrodynamic mechanical seals is not in contact with the mechanical seal type and withstands the pressure of the system in proportion by throttle, though each seal itself has the ability to withstand the total system pressure of the RCP [RCS].

There are two leak-off lines, one is the low pressure leakage line which collects the leakage of third stage seal and the other is high pressure leakage line which collects the leakage from the throttle. The leakage of the low pressure leakage line is connected to the RPE [NIVDS] and the high pressure leakage line is connected to the RCV [CVCS].

The standstill seal is a static seal, which is activated by nitrogen pressure to ensure the leaktightness of the shaft seal assembly when the Reactor Coolant Pump stops.

The seal injection water system consists of the seal injection water supplied by the RCV [CVCS] and emergency injection water from the casing.

The seal injection water supplied by the RCV [CVCS] is injected in to the seal housing.

The injection water pressure is slightly higher than the RCP [RCS] to avoid the reactor coolant getting outside from the RCP [RCS] due to seal leakage. One part of the injection water flows through the seal throttles and seal and the rest of the seal injection water flows downstream to lubricate the guide bearing, and finally mixes with the reactor coolant above the impeller.

The RRI [CCWS] supplies cooling water to the high pressure cooler. During normal operation, the high pressure cooler limits the temperature of injection water. If a loss of the seal injection water supplied by the RCV [CVCS] occurs, the reactor coolant in the casing will go up through the shaft, and the high pressure cooler will cool down the reactor coolant before it enters into the shaft seal assembly.

6.5.7.2.3 Motor

The motor is an asynchronous, squirrel cage, self-ventilated type motor, and can be started directly at full-voltage. The motor is composed of the lower guide bearing, rotor-stator assembly, axial thrust bearing, upper guide bearing, flywheel and the anti-rotation device. Each motor is fitted with two heat exchangers cooled by RRI [CCWS], on each side of the motor frame. The motor is equipped with heaters to protect the windings from condensation.

The guide bearings are a pad type bearing and the thrust bearing is comprised of a double thrust bearing. All bearings are oil lubricated. Water from the RRI [CCWS] feeds the external oil cooler for the upper guide bearing and the integrated oil cooler for the lower guide bearing. A high-pressure oil injection system provides sufficient pressure to establish an oil film between the thrust bearing pads to prevent wearing during start up and shutdown of the motor. The high-pressure oil injection system also sprays oil into the upper guide bearing. In order to reduce the risk of oil leakage and the likelihood for the outbreak of fire, oil collection devices are attached to the motor for all potential oil leak sources. Under normal operating conditions no oil shall be present in these oil collection devices.

The motors' internal components are cooled by air. Integrated fans at each end of the rotor draw air in through inlets in the frame of the motor. The air circulates within the motor, in particular to the stator end windings, and is then discharged through the heat exchangers into the pump room. The heat exchangers are designed to maintain the discharged air at an optimal temperature.

The flywheel fixed on the motor shaft can increase the moment of inertia of the component and can extend the Reactor Coolant Pump coast down time in case of LOOP.

The anti-rotation device installed on the motor prevents the Reactor Coolant Pump from reverse rotation.

6.5.7.2.4 Instrumentation

The operating parameters as well as operating status of the Reactor Coolant Pump shall be monitored and indicated to the operator.

The Reactor Coolant Pump is fitted with the following instrumentation to monitor the principal operating parameters:

a) 2 position detectors for the Standstill Seal System (SSS) (open and closed position);

b) 2 shaft displacement sensors at the level of the motor coupling sleeve;

c) 2 frame vibration sensors located on the lower flange of the motor;

d) 1 phase sensor with measurement targets;

e) 2 tachometric sensors for nominal speed measurement with measurement targets;

f) 1 tachometric sensor for the low speed measurement with measurement targets;

g) 2 temperature detectors at one location at the shaft seal inlet;

h) 2 temperature detectors for the motor lower guide bearing pads;

i) 2 temperature detectors for the motor upper guide bearing pads;

j) 2 temperature detectors for each bearing side of the thrust bearing;

k) 6 temperature detectors for motor stator windings (2 per phase);

l) 1 pressure gauge and 1 pressure switch on the oil injection device for each double thrust bearing;

m) 1 pressure transducer located on the cavities upstream of the second hydrodynamic seal stage;

n) 1 pressure transducer located on the cavities upstream of the third hydrodynamic seal stage;

o) 1 pressure transducer located in each seal leakage line;

p) 1 pressure transducer located in each seal water injection line;

q) 1 pressure transducer located on the SSS nitrogen storage tank;

r) 1 pressure transducer located on the SSS nitrogen discharge line;

s) 1 loose part detection system;

t) 1 flowmeter located in each seal water injection line;

u) 1 flowmeter located in each seal leakage line;

v) On each oil tank:

    1) 1 thermometer for temperature measurement;

    2) 1 high oil level and 1 low oil level detector providing on/off signals to

processing devices;

3) 1 sight level for the local verification of the oil level.

6.5.7.2.5 Supports

The Reactor Coolant Pump is supported vertically by three support columns fixed to three casing support lugs, and restrained horizontally by two snubbers fixed to the upper flange of the motor stand. The support schematic diagram is shown in Figure F-6D-16.

6.5.7.2.6 Equipment Insulation

The Reactor Coolant Pump is insulated with the cassette-type insulation filled with stainless steel foils (RMI type). The structure of Reactor Coolant Pump insulation is shown in Figure F-6D-17.

6.5.7.3 Design Principles and Codes

The Reactor Coolant Pump is designed according to recognised international codes and standards in accordance with OPEX. The Reactor Coolant Pump has a similar design compared to other Reactor Coolant Pumps in existing typical pressurised water reactors, and the Reactor Coolant Pump constructed from proven materials and manufacturing processes.

The design principles of the Reactor Coolant Pump are as follows:

a) The Reactor Coolant Pump design shall satisfy the safety functional requirements described in Sub-chapter 6.5.7.1;

b) The hydraulic parts shall be designed with the best estimated operating point;

c) The shaft seal leakage under a SBO condition shall be within an acceptable level within 24h;

d) The shaft seal and the lower guide bearing of the pump shall be maintained without dismantling the motor;

e) The Reactor Coolant Pump rotor shall be equipped with a flywheel to provide sufficient inertia;

f) The Reactor Coolant Pump shall coast down without thrust bearing oil injection, and the Reactor Coolant Pump shall not be damaged to lose its functionality after this situation.

For the pressure boundary of Reactor Coolant Pump, ASME code is selected as the code of design, fabrication, inspection and testing. Pre-service inspection and in-service Inspection of the Reactor Coolant Pump are performed according to the RSE-M code. The Reactor Coolant Pump casing safe ends connected to the Reactor Coolant Piping are compliant with RCC-M code. Relevant safety demonstration is provided in Structural Integrity safety cases (see PCSR Chapter 17).

6.5.7.4 Classification

According to the safety categorisation and classification presented in Reference [34] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of Reactor Coolant Pump is presented in Table T-6C-4.

6.5.7.5 Materials, Manufacturing and Inspection

6.5.7.5.1 Materials

Material selection for the Reactor Coolant Pump parts fully considers the engineering experience feedback and is in accordance with the requirements of applicable codes. All parts of the pump in contact with the reactor coolant are manufactured from stainless steel, except for the seals, bearings and special parts. The procurement and manufacture of the material shall meet the requirement of the applicable codes. The material grades of the main parts are as follows:

a) Casing: SA-508M Grade 3 Class 1;

b) Inlet and outlet nozzle safe ends: Z2 CND 18-12;

c) Impeller and diffuser: 1.4313 + V;

d) Pump shaft: 1.4313.09R;

e) Shaft seal housing: SA-336M Grade F6NM;

f) Flywheel: 27NiCrMoV 15-6.

The filler metal of Reactor Coolant Pump is selected according to ASME code and OPEX. The filler metal of the Reactor Coolant Pump is accepted according to ASME Section II, Part C and Section III NB-2400. Furthermore, some additional requirements above code compliance are specified for the filler metal which takes the manufacturing experience of previous Reactor Coolant Pumps into account.

Prohibited materials for internal and external surfaces are as follows:

a) Nitrided surfaces (does not apply to the motor);

b) Contaminants (does not apply to the motor);

c) Aluminium and aluminium alloys, except where approved by the Buyer prior to fabrication;

d) Antimony and silver are prohibited as main constituents of parts in contact with the primary coolant.

6.5.7.5.2 Manufacturing

Manufacturing of the Reactor Coolant Pump shall meet the requirements of ASME.

Besides, supplementary requirements from RCC-M or RSE-M codes are considered in Reactor Coolant Pump design and construction, which are confirmed as good practices/feedbacks. Liquid penetrated examination for the pump shaft shall be performed after the final machining of the shaft.

6.5.7.5.3 Inspection

PSI and ISI shall be performed according to the RSE-M code. For pressure boundary parts of the pump, the NDT shall be performed by qualified inspectors with qualification approved by certification.

6.5.7.5.4 Test

In order to evaluate the function of the Reactor Coolant Pump, the following major tests shall be carried out for Reactor Coolant Pump, more information is detailed in Reference [24]:

a) Full flow rate test:

    1) Hydraulic and mechanical performance test;

    2) Coast down test without thrust bearing oil injection;

    3) Losing RRI [CCWS] cooling water and/or RCV [CVCS] seal injection water performance test;

    4) Hydrostatic test;

b) Shaft seal assembly test:

    1) Shaft seal assembly performance test;

    2) SBO qualification test;

c) Motor test:

    1) Flywheel over speed test;

    2) Motor performance test.

6.5.7.6 Structural Integrity

The structural integrity of the Reactor Coolant Pump is demonstrated in Reference [40] in the form of CAE from a number of aspects, which mainly include applicable design codes and standards, loading conditions, design analysis, selected material, manufacture, manufacturing inspections, operation and maintenance.

6.5.8 Pressuriser Safety Valve

6.5.8.1 Safety Functional Requirements

The safety functional requirements of the PSVs are as outlined below:

a) During plant normal operation, the PSVs stay closed in order to:

   1) Prevent degradation of reactor coolant inventory to support the heat removal function;

   2) Serve as a pressure retaining boundary to prevent discharge of radioactive material.

b) Under DBC conditions which induce overpressure of the primary loop, the PSVs perform the overpressure protection functions via:

   1) Passive opening via a mechanical pilot under a hot overpressure condition;

   2) Active opening via a solenoid pilot under a cold overpressure condition as a backup of the safety valve of the RIS [SIS].

c) Under DEC-A conditions, the PSVs are activated and opened to perform the F&B function.

d) Under DEC-B conditions, there are no specific functional requirements' derived for the PSVs. However, they can still be passively opened via a mechanical pilot if the pressure of primary loop exceeds the opening set points of the PSVs.

e) The opening/closing status of the PSVs shall be monitored and indicated to the operator.

6.5.8.2 Description

Each PSV comprises a main valve, spring pilots and solenoid pilots. Each PSV equips actuators with two trains of spring-loaded pilot valves in parallel and one solenoid pilot valve; the two spring-loaded pilot valves are part of the design for redundancy. The main valve of the PSV can be opened by the activation of either one of the spring pilot valves or the solenoid pilot valve. In order to maintain the temperature of fluid in the PSVs and protect the safety of personnel, the PSVs are insulated with RMI. The general parameters of the PSVs are given in Table T-6C-15.

The PSVs will be designed and constructed by the equipment vender selected according to the technical specification which defines the requirements of design, material procurement, fabrication, inspection, examination and testing.

6.5.8.3 Design Principles and Codes

The PSVs are designed according to recognized international codes and standards with operating experience feedback. The PSVs are manufactured using proven materials and manufacturing processes.

For the PSVs, the RCC-M code is selected as the code of design, fabrication, inspection and testing. Pre-Service Inspection and In-Service Inspection of the PSVs shall follow the RSE-M code.

The design principles for the PSVs are as following:

a) The PSVs design shall satisfy the safety functional requirements described in Sub-chapter 6.5.8.1;

b) The actuator shall be of a pilot operated type with redundancy designed in, and can prevent spurious operation;

c) All pilot valves shall take pressure from the main valve. Pressure sensing line from the PZR or upstream pipe of the main valve is not preferred;

d) The PSVs design shall satisfy the operating conditions (such as discharge flowrate, opening and closing stroke time, dead time of valve opening and closing) described in Sub-chapter 6.4.2.2.3.

6.5.8.4 Classification

According to the safety categorisation and classification presented in Reference [8] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of the PSV is presented in Table T-6C-4.

6.5.8.5 Materials, Manufacturing and Inspection

6.5.8.5.1 Materials

Material selection of the PSVs fully considers engineering experience feedback and is in accordance with the requirements of applicable codes. The pressure retaining parts of the isolation valves are made of stainless steel and the internal components are selected from corrosion resistant materials. Applicable procurement specifications for the valve materials are in accordance with RCC-M B2200.

6.5.8.5.2 Manufacturing and Inspection

The manufacturing and inspection of the pressure retaining parts shall be implemented in accordance with the relative provisions of the RCC-M.

6.5.8.5.3 Qualification and Testing

In order to assess the function of the PSVs, the PSVs shall be qualified according to their related qualification requirements. The following major tests or analysis shall be carried out, relevant qualification schedule is detailed in Reference [24]:

a) Seal performance test;

b) Discharge capacity test;

c) Operability test;

d) Cyclic life test;

e) Vibration ageing test;

f) Seismic test;

g) Environmental ageing test.

6.5.9 Severe Accident Dedicated Valves

6.5.9.1 Safety Functional Requirements

The safety functional requirements of the SADVs are as stated below:

a) During plant normal operation, or under DBC and DEC-A conditions, the SADVs stay closed in order to:

  1) Prevent degradation of the reactor coolant inventory to support the heat removal function;

  2) Serve as a pressure retaining boundary to prevent the discharge of radioactive material.

b) Under DEC-B conditions, the SADVs perform a fast depressurisation function to avoid high pressure core melt;

c) The opening/closing status of valves shall be monitored and indicated to the operator.

6.5.9.2 Description

The SADVs are designed as two trains in parallel to increase the reliability under DEC-B condition. Each train has two isolation valves, one is motor-driven gate valve, and the other is motor-driven globe valve. Water block sealing is designed upstream of the gate valve to improve the reliability of the valve opening.

Under normal operation and design basis accidents, the SADVs remain closed. Under severe accidents when the temperature at the inlet of the valve reaches to 600°C, the SADVs have the capability to be opened manually.

The general parameters of the SADVs are given in Table

T-6C-16. The general arrangement is shown in Figure F-6D-18.

The SADVs will be detailed designed and constructed by the equipment vender selected according to the technical specification which defines the requirements of the design, material procurement, fabrication, inspection, examination and testing.

6.5.9.3 Design Principles and Codes

The SADVs are designed according to recognised international codes and standards with operating experience feedback. The SADVs are constructed from proven materials and manufacturing processes.

For the SADVs, the RCC-M code is selected as the code of design, fabrication, inspection and testing. Pre-Service Inspection and In-Service Inspection of the SADVs shall follow the RSE-M code.

The design principles for the SADVs are as follows:

a) Under normal and design basis conditions, the SADVs shall be closed and leak tight;

b) Under severe accident conditions, the SADVs shall be opened at a required high temperature. When the SADVs are open, the valves shall never be closed again;

c) The SADVs design shall satisfy the discharge flowrate described in Sub-chapter 6.4.2.2.3.

6.5.9.4 Classification

According to the safety categorisation and classification presented in Reference [8] as well as the method and requirements of structural integrity classification presented in Reference [32], the classification of the SADV is presented in Table T-6C-4.

6.5.9.5 Materials, Manufacturing and Inspection

6.5.9.5.1 Materials

Material selection of the SADVs fully considers the engineering experience feedback and is in accordance with the requirements of applicable codes. The pressure retaining parts of the isolation valves are made of stainless steel and the internal components are selected from corrosion resistant materials. Applicable procurement specifications for the valve materials are according to RCC-M B2200 and RCC-MR.

6.5.9.5.2 Manufacturing and Inspection

The manufacturing and inspection of the pressure retaining parts shall be implemented in accordance with the relative provisions of the RCC-M.

6.5.9.5.3 Qualification and Test

In order to assess the function of the SADVs, the SADVs shall be qualified according to the related qualification requirement. The following major tests or analysis shall be carried out, relevant qualification schedule is detailed in Reference [24]:

a) Sealing capability test;

b) Flow coefficient test;

c) Operability test;

d) End loading test;

e) Cyclic life test;

f) Vibration ageing test;

g) Seismic test;

h) Flow interruption test;

i) Environmental ageing test.

6.5.10 Isolation Valves

6.5.10.1 Safety Functional Requirements

The isolation valves presented in this sub-chapter are commonly used valves which constitute the pressure retaining boundary.

These valves mainly perform the following safety functions by providing reliable isolation:

a) Maintain the reactor coolant inventory to support the heat removal function;

b) Serve as a barrier to confine the radioactive effluent.

6.5.10.2 Description

These isolation valves mainly include gate valves, globe valves and check valves.

Gate valves are fitted with packing seals and leak-off line.

Globe valves are fitted with stem bellow seals in order to provide total external leaktightness, and packing seals are also used with leak-off lines when the valve nominal diameter is larger than 50 mm.

Swing type check valves have no penetration parts across the valve body.

In order to maintain the temperature of the fluid in the isolation valves and protect the safety of personnel, the isolation valves which are located in Reactor Building are insulated with RMI.

6.5.10.3 Design Principles and Codes

The isolation valves are designed according to recognised international codes and standards with operating experience feedback. The isolation valves are manufactured from proven materials and manufacturing processes.

For the isolation valves, RCC-M code is selected as the code of design, fabrication, inspection and testing. PSI and ISI of the isolation valves shall follow the RSE-M code.

6.5.10.4 Classification

According to the safety functions which are performed by these valves as well as the function category and design provision category, the classification of these isolation valves are presented in Table T-6C-4.

6.5.10.5 Materials, Manufacturing and Inspection

6.5.10.5.1 Materials

Material selection of the isolation valves fully considers engineering experience feedback and is in accordance with the requirements of applicable codes. The pressure retaining parts of the isolation valves are made of stainless steel and the internal components are selected from corrosion resistant materials. Applicable procurement specifications for the valve materials are according to RCC-M B2200.

6.5.10.5.2 Manufacturing and Inspection

The manufacturing and inspection of the pressure retaining parts shall be implemented in accordance with the relative provisions of the RCC-M.

6.5.10.5.3 Qualification and Test

In order to assess the function of the gate valves and globe valves, the following tests or analysis shall be carried out:

a) Sealing capability test;

b) Flow coefficient test;

c) Operability test in extreme conditions;

d) End loading test;

e) Cyclic life test;

f) Vibration ageing test;

g) Operability test in accident conditions;

h) Seismic test or analysis;

i) Flow interruption test;

j) Particles test (if required).

In order to assess the function of the check valves, the following tests or analysis shall be carried out:

a) Sealing capability test;

b) Operability test in extreme conditions;

c) Flow coefficient test;

d) Movement performance test;

e) End loading test;

f) Cyclic life test;

g) Vibration ageing test;

h) Operability test in accident conditions;

i) Seismic test or analysis;

j) Flow interruption test;

k) Particles test (if required);

l) Environmental ageing test.

6.6 Description of Overpressure Protection

Overpressure is defined as when the primary or secondary pressure exceeds the Design Pressure (DP) of the system. It may be caused by thermal unbalance or excessive fluid injection. In overpressure conditions, the integrity of system is challenged. Therefore, overpressure protection is essential to protect the integrity of primary and secondary systems. For the UK HPR1000, overpressure can be mitigated by PSVs, Main Steam Safety Valve (MSSVs), VDA [ASDS], GCT [TBS] and pressuriser spray, as well as the reactor protection system.

The design of the primary and secondary systems and components are mainly based on the provisions of RCC-M, so the methodology and principles of overpressure analysis are consistent with RCC-M.

Overpressure analysis shall be evaluated in the following conditions to ensure the completeness of the overpressure protection:

a) Overpressure analysis in hot conditions;

b) Overpressure analysis in cold shutdown conditions.

For overpressure analysis in hot conditions, the purpose is to demonstrate that the integrity of the primary and secondary pressure boundary is ensured.

According to RCC-M, three categories are classified with respect to the overpressure conditions:

a) Category 2: Normal operating conditions and upset conditions, corresponding to DBC-1 and DBC-2 transients;

b) Category 3: Emergency conditions, corresponding to DBC-3 events;

c) Category 4: Conditions which are highly improbable, corresponding to DBC-4 events and multiple event sequences.

The acceptance criteria adopted in overpressure analysis are consistent with the design of the primary and secondary systems. The acceptance criteria are as follows:

a) Category 2: The maximum pressure shall not exceed 100% DP;

b) Category 3:

1) The maximum pressure shall not exceed 110% DP if all safety valves are available;

2) The maximum pressure shall not exceed 120% DP if one safety valve is unavailable.

c) Category 4: The maximum pressure shall not exceed 130% DP.

In the overpressure analysis, the transients related to primary and secondary side overpressure are identified for each category. Meanwhile, the conservative initial conditions and boundary conditions are taken into account to maximise the peak pressure in the primary and secondary side.

The detailed analyses of overpressure protection at power are presented in References [41], [42], [44], [45], [46] and [47]. The results show that acceptance criteria presented above are met, the primary and secondary pressure boundary integrity are protected in the overpressure transients.

In cold shutdown state, especially in the water solid state (i.e. RCP [RCS] and PZR are full of water), the RPV could be under the risk of brittle fracture at low RCP [RCS] temperature and high RCP [RCS] pressure. Therefore, for overpressure analysis in cold shutdown conditions, the object is to protect RPV from brittle fracture.

In cold shutdown state, overpressure is caused by thermal unbalance or excessive fluid injection. The transients with overpressure risk are classified in two types:

a) Transients leading to RCP [RCS] inventory increase;

b) Transients leading to RCP [RCS] energy increase.

According to RCC-M rules, the acceptance criteria of overpressure in cold shutdown state are as follows:

a) Category 2: The maximal RCP [RCS] pressure shall not exceed the limitative pressure (To avoid a risk of brittle fracture of the RPV, the allowable pressure of RPV corresponding to the minimum temperature of RCP [RCS] during the overpressure transient is considered as the limitative pressure) related to RPV brittle facture.

   For the category 2 transients, the RHR design pressure (6.4 MPa), which is lower than the limitative pressure related to RPV brittle facture, is conservatively selected as the final acceptance criteria.

b) Category 3: The maximal RCP [RCS] pressure shall not exceed the limitative pressure related to RPV brittle facture.

c) Category 4: The maximal RCP [RCS] pressure shall not exceed the limitative pressure related to RPV brittle facture.

The detailed analyses of overpressure protection in cold shutdown conditions are presented in References [48], [49] and [50]. The results show that acceptance criteria presented above are met, the RPV is protected from brittle fracture during the overpressure transients in cold shutdown state.

6.7 ALARP Assessment

The ALARP demonstration on RCP [RCS] is undertaken in line with the general ALARP methodology for the UK HPR1000. The overall strategy/approach of ALARP demonstration is introduced in *ALARP Methodology*, Reference [51]. The simplified flow chart is presented in Figure F-6D-19.

6.7.1 Holistic ALARP Assessment

6.7.1.1 Evolution of Reference Design

RCP [RCS] of reference plant is a typical 3-Loops system developed from M310, through the CPR1000, CPR1000+, ACPR1000 and then became the reference plant. RCP [RCS] is mature and the technology used in RCP [RCS] is proven. The RCP [RCS] of UK HPR1000 has been adopted from the reference plant. A historic review for RCP [RCS] is carried out. The overall evaluation of the reference plant is introduced in PCSR Chapter 2 and *HPR1000 R&D History*, Reference [52].

The design of the RCP [RCS] ensures the reactivity control provided by the reactor coolant water, the heat removal from the core to the secondary cooling side via reactor coolant, and the confinement of radioactive material. The design of the RCP [RCS] has been improved based on the ACPR1000 design with due consideration of proven techniques and the wide experience from the design, manufacture, construction, commissioning and operational feedback from existing NPPs. The following examples are the major modifications implemented during the development of the HPR1000 design:

a) Increase in the volume of the Pressuriser (PZR). The PZR controls the pressure of the RCP [RCS] during normal operation and transient operation of the NPP. By increasing the volume of the PZR, the pressure control ability is improved, which helps reduce the risk of overpressure and maintain the integrity of the primary pressure boundary.

b) Increase in the volume of secondary side Steam Generator (SG). SGs serve as the first means for heat removal from the reactor. By increasing the secondary side volume in the SGs, the inherent safety of the HPR1000 can be improved. The larger volume of the SGs contributes to improving the ability of temperature control and improving the autonomy of the SGs. For example, in SGTR conditions, the larger secondary side steam volume of the SGs can prevent the overfill of the affected SG, which improves the resilience to transients and accidents.

c) Eliminating the penetrations in the RPV lower head. The RPV is the highest reliability pressure boundary, and contains the reactor core, core support structure and water coolant. In the ACPR1000 RPV design, measurement instrumentation adapters penetrate the lower head of the RPV, which may challenge the integrity of the pressure boundary and may increase the risk of leakage from the lower head. By considering international research feedback, lower head penetrations are eliminated in the HPR1000 design and the measurement instrumentation adapters are implemented on the closure head on top of the RPV, which improves the integrity of the highest reliability pressure boundary.

More information related to the design evaluation of RCP [RCS] is presented in *ALARP Demonstration for Reactor Coolant System*, Reference [6]. The information shows that the design of the RCP [RCS] follows the ALARP way to moving forward.

6.7.1.2 Compliance with RGP

During UK HPR1000 GDA, CGN developed its working strategy on the RGP compliance analysis, and recorded in the *ALARP Demonstration for Reactor Coolant System*, Reference [6]. The optimised strategy outlines the process that is adopted to ensure that the analysis forms a comprehensive basis for the ME design.

According to the working strategy mentioned above, a group of applicable RGPs are identified for the SSCs of the RCP [RCS] (e.g. *The safe isolation of plant and equipment* - HSG253, Reference [43]). The detailed applicable analysis and the applicable RGP list is presented in *Suitability Analysis of RGP for Sample of Dynamic SSC*, Reference [53] and *Suitability Analysis of RGP for Sample of Static SSCs*, Reference [54], and then summarised in Reference [6]. The main codes and standards used in the SSC design of the RCP [RCS] are presented in the Table T-6C-2 and Reference [6]. A complete list of RGPs that applicable is presented in Reference [6].

Detailed compliance analysis is carried out during the GDA stage. The strategy, method, and the working process are introduced in Reference [6]. Several gaps are identified and recorded in Reference [6]. Optioneering is carried out for the gaps identified to reduce the risk level within UK HPR1000 RCP [RCS] design. Following the decision making process, the ALARP options for these gaps are selected for UK HPR1000 RCP [RCS] and integrated in the design reference document (i.e. the SDMs).

6.7.1.3 OPEX Review

During GDA, CGN developed and optimised its methodology for the identification and utilisation of valuable OPEX. The developed methodology is provided in Reference [6]. The overview of the OPEX approach comprises eight steps as below:

a) Step 1: Identification of potential sources of OPEX;

b) Step 2: Collection of OPEX;

c) Step 3: Identification of topics/themes;

d) Step 4: Prioritisation process;

e) Step 5: Determination of the intended use;

f) Step 6: Screening process;

g) Step 7: Use of OPEX;

h) Step 8: Justification of OPEX.

During GDA, an expectation is derived from the UK context related to the scope of Reflective Metallic Insulations (RMI) used in the primary loop. This is further discussed in Reference [6].

According to the lesson learned from RGP and OPEX, the use of RMI for insulation is becoming more widespread. Compared to the fibreglass insulation material, the RMI provides several benefits; therefore, CGN carried out the optioneering process to support the insulation material selection. According to the result of the optioneering on insulation material selection, all the glass fibre insulation material in containment needed to be replaced by RMI. The further ALARP demonstration is introduced in Reference [6] and relevant lower tier supporting safety case documents.

6.7.1.4 Risk Assessment Insight

A group of safety cases which providing the risk insight are produced and recorded the potential area to be enhanced, within various risk assessment technical areas. The information below presents the summary of the potential risk related to RCP [RCS] to the plant and workers, derived from the risk estimate areas. The risk is summarised in in Reference [6].

6.7.2 Specific ALARP Assessment

6.7.2.1 Summary of Optioneering

From the holistic ALARP assessment, several potential improvements to the design are identified and optioneered. The associated analysis report has been produced and the system design manuals modified. The Optioneering of the Reactor Coolant System is summarised below, with more detailed information presented in Reference [6].

6.7.2.2 Reflective Metallic Insulations

Even though the RGPs investigation report recommends the RMI as the only insulation material in containment, other factors related to the debris effect such as the nuclear safety, personnel health protection, cost, decommissioning etc. still needed to be considered, Therefore, CGN carried out an optioneering study to determine the upstream material selection related to the debris inside the containment.

According to the result of the optioneering study, all the glass fibre insulation material in containment is replaced by RMI. More detailed information is discussed in Reference [6].

6.7.2.3 Works Involving Valve Inspection and Maintenance

Valves inspection and maintenance is one of the major activities carried out during plant operation to ensure that all the valves, especially those playing an important safety function, are maintained in good performance. This activity comes with high radiological risk mainly due to the large amount of irremovable valves needing to be inspected and maintained.

An ALARP study for occupational exposure associated with valve inspection and maintenance has been carried out by means of relevant RGP compliance analysis and ERIC/PPE review following the hierarchy of control philosophy.

According to the outcome of the ALARP study, the risk of occupational exposure due to valve inspection and maintenance is concluded to be ALARP. More detailed information is discussed in Reference [6].

6.7.2.4 Works Involving RPV

The works involving RPV are those activities related to the RPV carried out during outage to ensure that the reactor core can work continuously as anticipated during the following fuel cycle(s), which includes mainly the RPV head assembly lifting and the ICIA replacement. This activity comes with high radiological risk due to the high radiation level of the RPV. An ALARP study has been carried out for these activities, as follows:

a) RPV head assembly lifting;

b) ICIA Replacement.

According to the outcome of the ALARP study, the risk of occupational exposure due to works involving RPV is concluded to be ALARP. More detailed information is discussed in Reference [6].

6.7.2.5 Works Involving SG

The work involving SG (i.e. SG inspection and maintenance) is one of the activities with high dose risk mainly due to the high radiation level of SG. An ALARP study for occupational exposure associated with work involving the SG has been carried out by means of RGP compliance analysis, OPEX review and application of ERIC/PPE following hierarchy of control philosophy.

According to the outcome of the ALARP study, the risk of occupational exposure due to SG inspection and maintenance can be considered to be ALARP. More detailed information is discussed in Reference [6].

6.7.3 ALARP Assessment Conclusion

According to the information presented above, which is in line with the ALARP demonstration approach, it is concluded that there is no significant gap/shortfall in the RCP [RCS] design that can impact the fundamental safety of the UK HPR1000. The design of the RCP [RCS] is ALARP.

6.8 Concluding Remarks

This chapter, and its supporting references provide a robust safety demonstration that the Reactor Coolant System [RCS] for the UK HPR1000 has been designed to the high reliability consistent with their safety role for the UK HPR1000.

This chapter is a Tier 1 document and forms part of the overarching safety case for the UK HRP1000. This chapter presents the decoupled sub-claims and arguments that are used to support the fundamental objective of the UK HPR1000 mentioned in PCSR Chapter 1 and Section 6.2 of this Chapter, supported with a proportionate level of information derived from the Tier 2 supporting documents. The ALARP demonstration is based on the latest consolidated design scheme of RCP [RCS], i.e. Design Reference 3.0.

This chapter presents the route map in the form of table (Appendix 6A), which sets out a "direction of moving forward" for the RCP [RCS], clearly identifies the information important to safety, and points to the relevant evidences used to support the ALARP demonstration and design of RCP [RCS]. All the information and safety cases form comprehensive trail of Claim-Argument-Evidence.

Sub-chapter 6.2 presents the basic introduction to the PCSR Chapter 6, including the introduction of the claim, sub-claim and argument, the introduction of the document structure of this chapter, and the interfaces with other Tier 1 safety cases.

Sub-chapter 6.3 presents summarised information on the major Codes and Standards used in RCP [RCS] design, links to the ALARP demonstration and useful lower tier documents supporting the safety case, show the start point of the ALARP demonstration - from Relevant Good Practices.

Sub-chapter 6.4 introduces important design information on the RCP [RCS] at system/configuration level. This sub-chapter starts from the important safetyfunctional requirements and design bases of the design, and then presents the key summarised information to show that the design requirements and engineering principles are substantiated.

Sub-chapter 6.5 introduces important design information on the RCP [RCS] at component level. This sub-chapter is similar with sub-chapter 6.4, starts from the important safety functional requirements and design bases of the design, and then presents the key summarised design information. The structural integrity of the primary loop equipment is important to safety. The relevant ALARP demonstration is integrated in PCSR Chapter 17. So, only simplified summarised information is presented in Sub-chapter 6.5 and then linked out to PCSR Chapter 17.

Overpressure protection is important for the primary circuit of the NPP. Sub-chapter 6.6 presents the summarised information (for completeness) of the overpressure protection of UK HPR1000, covers the categories of the overpressure conditions, the acceptance criteria for these conditions, and key supporting safety cases.

Sub-chapter 6.7 presents import information on ALARP demonstration for RCP [RCS]. This sub-chapter presents the following key information:

a) Holistic ALARP Assessment

1) Lookback on the RCP [RCS] evolution of the reference design

This shows that the RCP [RCS] of the Reference Plant was developed following a “Risk-Based” and “Proportionate Approach”. The development of the RCP [RCS] meets the fundamental expectation of the ALARP principle. The relevant information is introduced in sub-chapter 6.7.1.1.

2) Compliance with RGP

It was observed that the RGPs used in UK are different to those applicable in China. Compliance with local RGP is an important step to ensuring that the design of the RCP [RCS] can meet the fundamental safety expectation in the UK.Accordingly, an analysis was performed to identify gaps and appropriate actions taken to mitigate the gaps. The relevant information is introduced in sub-chapter 6.7.1.2.

3) OPEX review

Large amounts of OPEX data from the global nuclear industry area was used to support the design and development of the UK HPR1000.

The methodology for using of OPEX is introduced and then the OPEX applicable to the RCP [RCS] is identified, such as RMI, water hammer, etc. This shows that OPEX is taken into account in the development of the UK HPR1000 RCP [RCS]. The relevant information is introduced in sub-chapter 6.7.1.3.

4) Risk Assessment

Besides the RGP and OPEX, potential shortfalls can be identified by various risk assessments, in the form of holistic risk review, such as PSA, Fault Studies, Radiological Protection, etc. A high level summary of relevant information is presented in sub-chapter 6.7.1.4 and supported by the ALARP demonstration report, Reference [6].

b) Specific ALARP Assessment

For the potential gaps identified from the risk assessment, specific ALARP assessments are carried out. The results of the ALARP assessments are used to inform design improvement or to demonstrate that the current design represents an optimised solution and no further proportionate design improvement can be made.

Sub-chapter 6.7.2 presents summary information showing the great effort made to ensure that the risk to UK persons associated with the RCP [RCS] design is reduced to ALARP.

During the different stages of GDA, forward actions to carry out specified safety assessment, or prepare relevant safety analyses were identified. These forward actions have all been completed at the end of GDA.

Finally, the information and the conclusion presented in PCSR Chapter 6 can support the following Claim(s) which are presented in Appendix 6A:

Claim 3.3.2: The design of the Reactor Coolant System has been substantiated.

a) *Sub-Claim 3.3.2.SC06.1: The safety functional requirements (Design Basis) have been derived for the system;*

b) *Sub-Claim 3.3.2.SC06.2: The system design satisfies the safety functional requirements;*

c) *Sub-Claim 3.3.2.SC06.3: All reasonably practicable measures have been adopted to improve the design;*

d) *Sub-Claim 3.3.2.SC06.4: The system performance will be validated by suitable commissioning and testing;*

e) *Sub-Claim 3.3.2.SC06.5: The effects of ageing of the system have been addressed in the design and suitable examination, inspection, maintenance and testing specified.*

According to the information above, a final conclusion can be made - *There is no significant gap/shortfall in the RCP [RCS] design that can impact the fundamental safety of the UK HPR1000. The risk level of RCP [RCS] by design is reduced to ALARP.*

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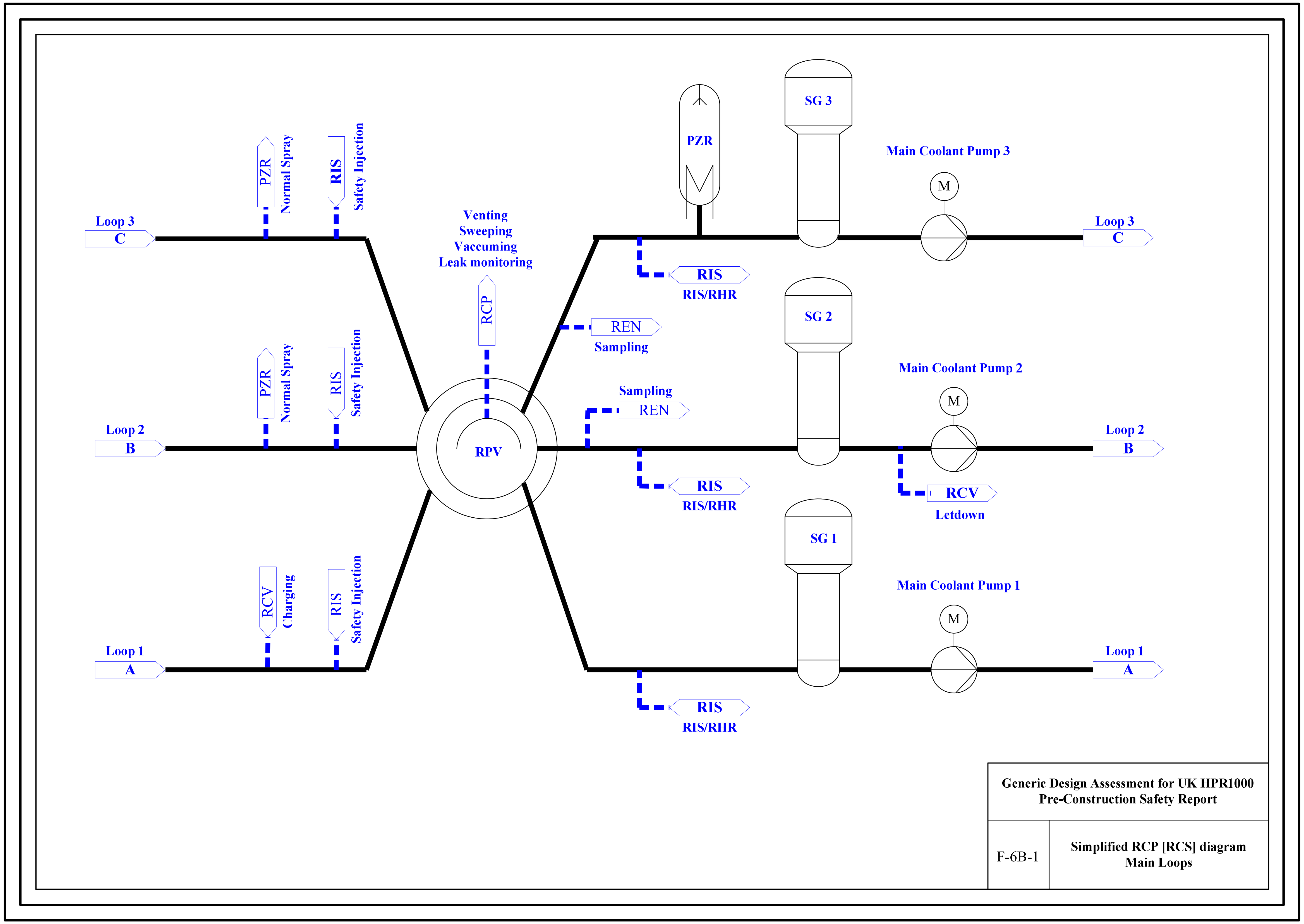
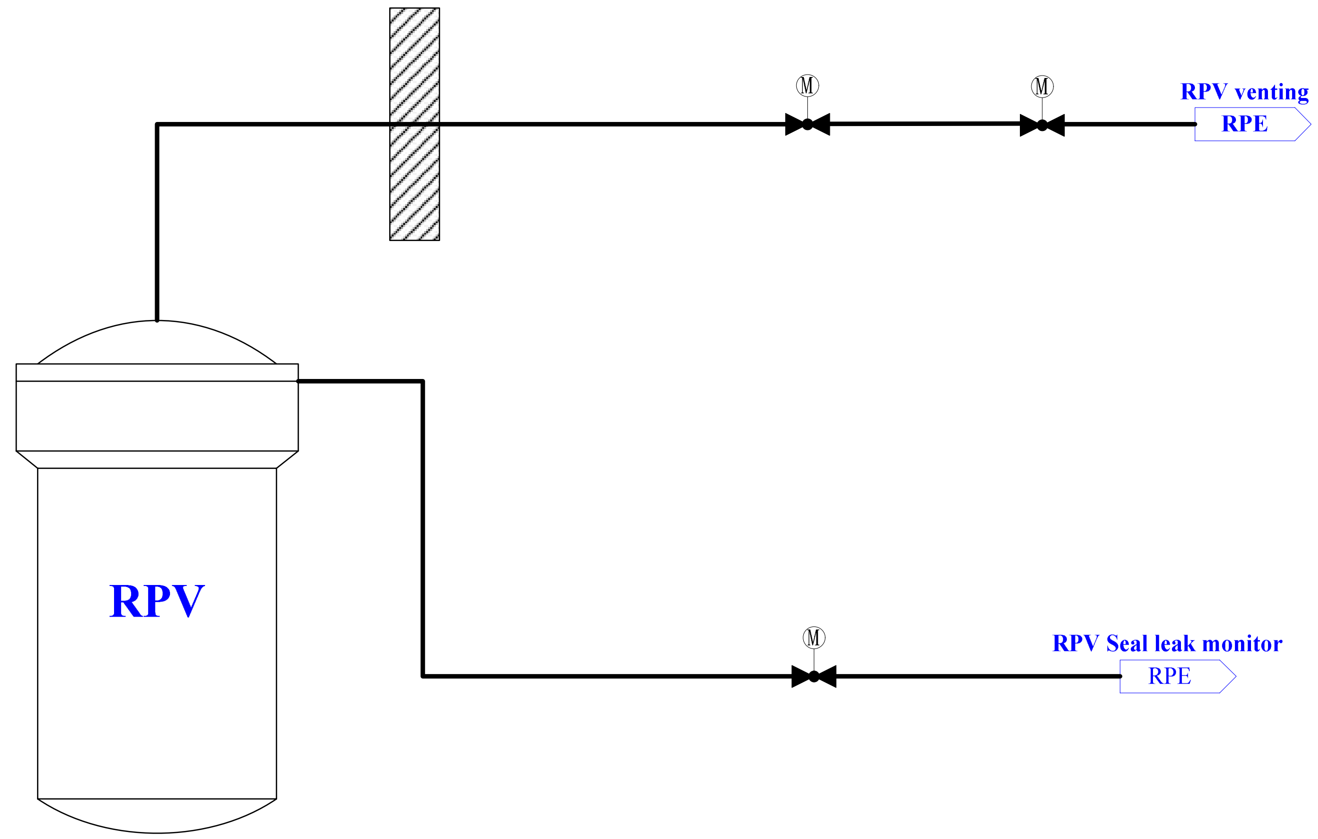
Appendix 6A Route Map

Claim		Sub-claim		Argument		PCSR Links	Evidences	
3.3.2	The design of the Reactor Coolant System has been substantiated.	3.3.2.SC06.1	The safety functional requirements (Design Basis) have been derived for the system.	3.3.2.SC06.1-A1	The specific design principles are identified for the Structures, Systems and Components (SSC) based on relevant good practice.	Sub-chapter 6.3 Sub-chapter 6.5.x.3	3.3.2.SC06.1-A1-E1 3.3.2.SC06.1-A1-E2 3.3.2.SC06.1-A1-E3	SDM Chapter 3 SDM Chapter 5 RGP Suitability Analysis Report
				3.3.2.SC06.1-A2	The design basis (requirements) of the SSC has been derived from the safety analysis in accordance with the general design and safety principles.	Sub-chapter 6.4.1 Sub-chapter 6.5.x.3	3.3.2.SC06.1-A2-E1 3.3.2.SC06.1-A2-E2 3.3.2.SC06.1-A2-E3	Engineering Schedule SDM Chapter 3 SDM Chapter 5
				3.3.2.SC06.1-A3	The Safety Class of the SSC has been identified from the safety analysis.	Sub-chapter 6.4.4 Sub-chapter 6.5.x.4	3.3.2.SC06.1-A3-E1 3.3.2.SC06.1-A3-E2	SDM Chapter 3 System Classification List
		3.3.2.SC06.2	The system design satisfies the safety functional requirements.	3.3.2.SC06.2-A1	Appropriate design methods have been identified for the SSC including design codes and standards.	Sub-chapter 6.4.1 Sub-chapter 6.5.x.3	3.3.2.SC06.2-A1-E1	SDM Chapter 3
				3.3.2.SC06.2-A2	The SSC have been analysed using the appropriate design methods and meet the design basis requirements.	Sub-chapter 6.4.4 Sub-chapter 6.5	3.3.2.SC06.2-A2-E1 3.3.2.SC06.2-A2-E2 3.3.2.SC06.2-A2-E3	SDM Chapter 4 SDM Chapter 6 SDM Chapter 9
				3.3.2.SC06.2-A3	The SSC analysis recognises interface requirements and effects from/to the interfacing SSC.	Sub-chapter 6.4.3 Sub-chapter 6.5	3.3.2.SC06.2-A3-E1	SDM Chapter 4
		3.3.2.SC06.3	All reasonably practicable measures have been adopted to improve the design.	3.3.2.SC06.3-A1	The SSC meet the requirements of the relevant design principles (generic and system specific) and therefore of relevant good practice.	Sub-chapter 6.4.4 Sub-chapter 6.5	3.3.2.SC06.3-A1-E1 3.3.2.SC06.3-A1-E2 3.3.2.SC06.3-A1-E3 3.3.2.SC06.3-A1-E4 3.3.2.SC06.3-A1-E5 3.3.2.SC06.3-A1-E6	SDM Chapter 3 SDM Chapter 4 SDM Chapter 6 RGP Compliance Analysis Report Equipment Specification SDM Chapter 9
				3.3.2.SC06.3-A2	PSA indicates the SSC are not disproportionate contributor to risk.	Sub-chapter 6.6	3.3.2.SC06.3-A2-E1	ALARP Reports from other areas
				3.3.2.SC06.3-A3	Design improvements have been considered in the SSC and any reasonably practicable changes implemented.	Sub-chapter 6.6	3.3.2.SC06.3-A3-E1	ALARP Demonstration for RCP [RCS]
		3.3.2.SC06.4	The system performance will be validated by suitable commissioning and testing.	3.3.2.SC06.4-A1	The SSC have been designed to take benefit from a suite of pre-construction tests, to provide assurance of the initial quality of the manufacture.	Sub-chapter 6.4.4 Sub-chapter 6.5.x.5	3.3.2.SC06.4-A1-E1 3.3.2.SC06.4-A1-E2 3.3.2.SC06.4-A1-E3 3.3.2.SC06.4-A1-E4	SDM Chapter 4 SDM Chapter 6 Equipment Qualification Requirements EQ Schedule

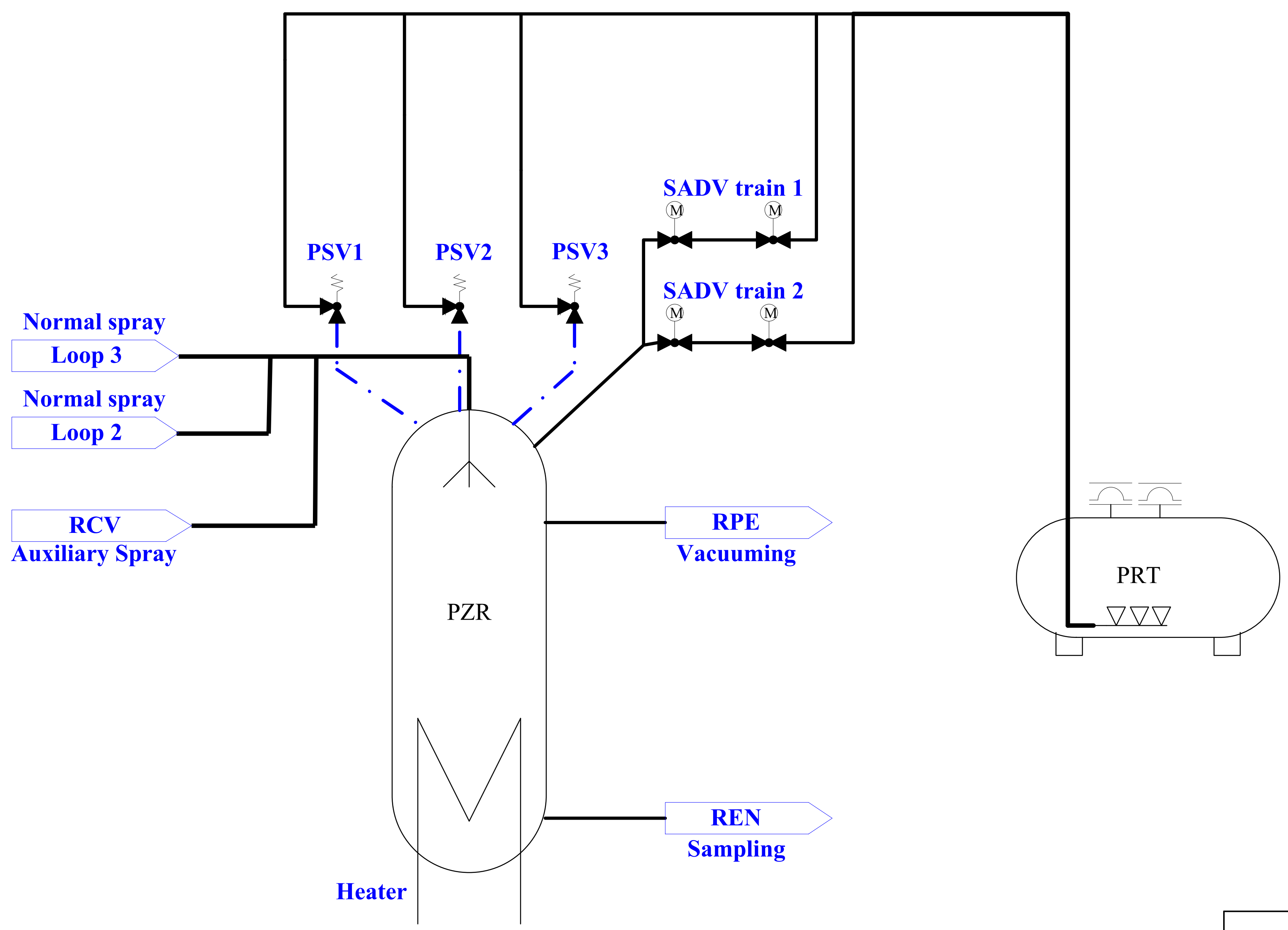
Claim	Sub-claim	Argument	PCSR Links	Evidences
		3.3.2.SC06.4-A2 The SSC has been designed to take benefit from a suite of commissioning tests, to provide assurance of the initial quality of the build.	Sub-chapter 6.4.4	3.3.2.SC06.4-A2-E1 SDM Chapter 6 3.3.2.SC06.4-A2-E2 System Commissioning Programme 3.3.2.SC06.4-A2-E3 EQ Schedule
	3.3.2.SC06.5 The effects of ageing of the system have been addressed in the design and suitable examination, inspection, maintenance and testing specified.	3.3.2.SC06.5-A1 An initial Examination, Maintenance, Inspection and Testing (EMIT) strategy has been developed for the SSC that are expected to be examined, maintained, inspected and tested.	Sub-chapter 6.4.4	3.3.2.SC06.5-A1-E1 SDM Chapter 6 3.3.2.SC06.5-A1-E2 Periodic Test Completeness Note 3.3.2.SC06.5-A1-E3 Pre-service Inspection List
		3.3.2.SC06.5-A2 The SSC that cannot be replaced have been shown to have adequate life, which includes the requirements during decommissioning.	Sub-chapter 6.4.4 Sub-chapter 6.5	3.3.2.SC06.5-A2-E1 SDM Chapter 6 3.3.2.SC06.5-A2-E2 Decommissioning Area Safety Case

Appendix 6B Functional Diagrams

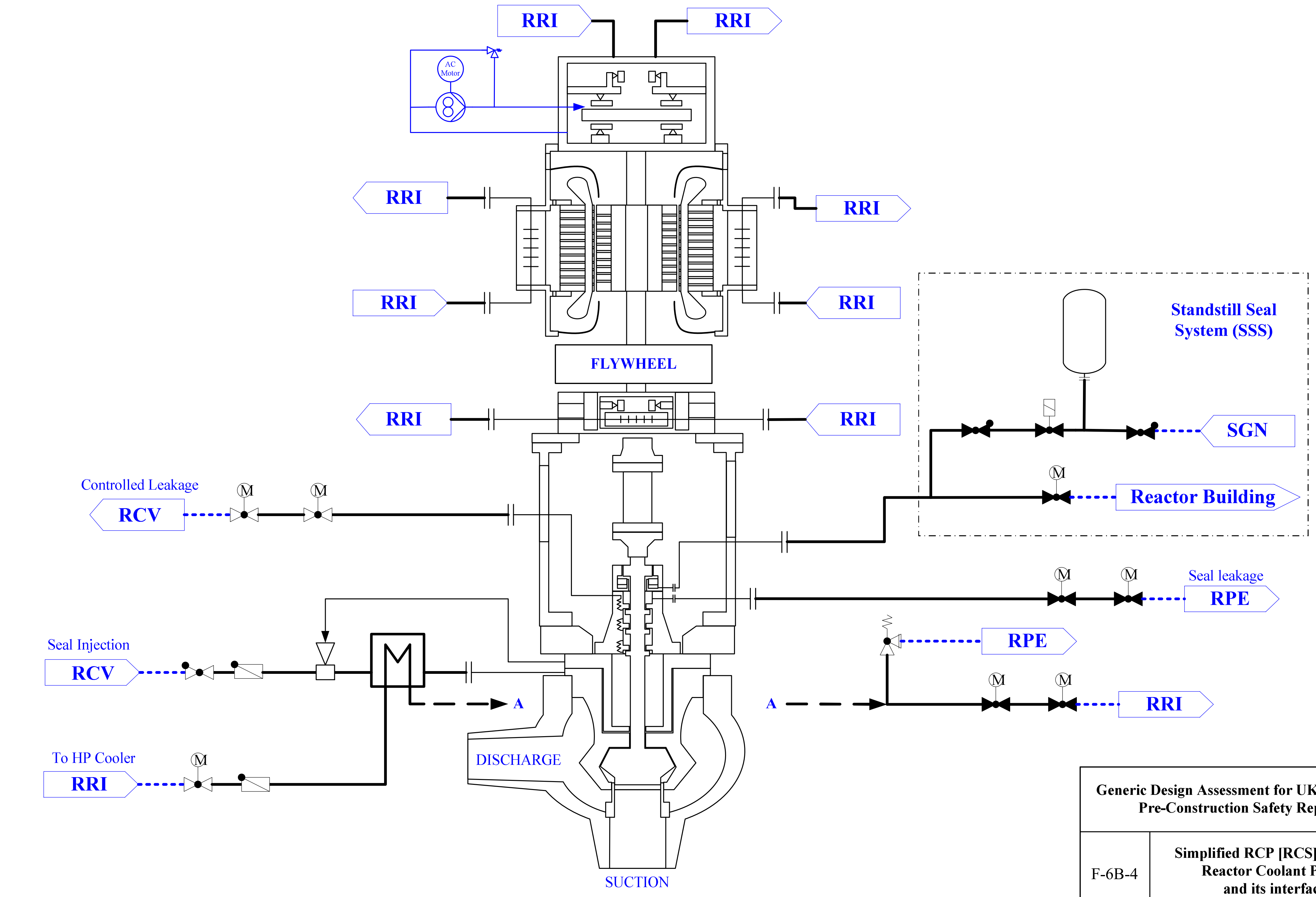




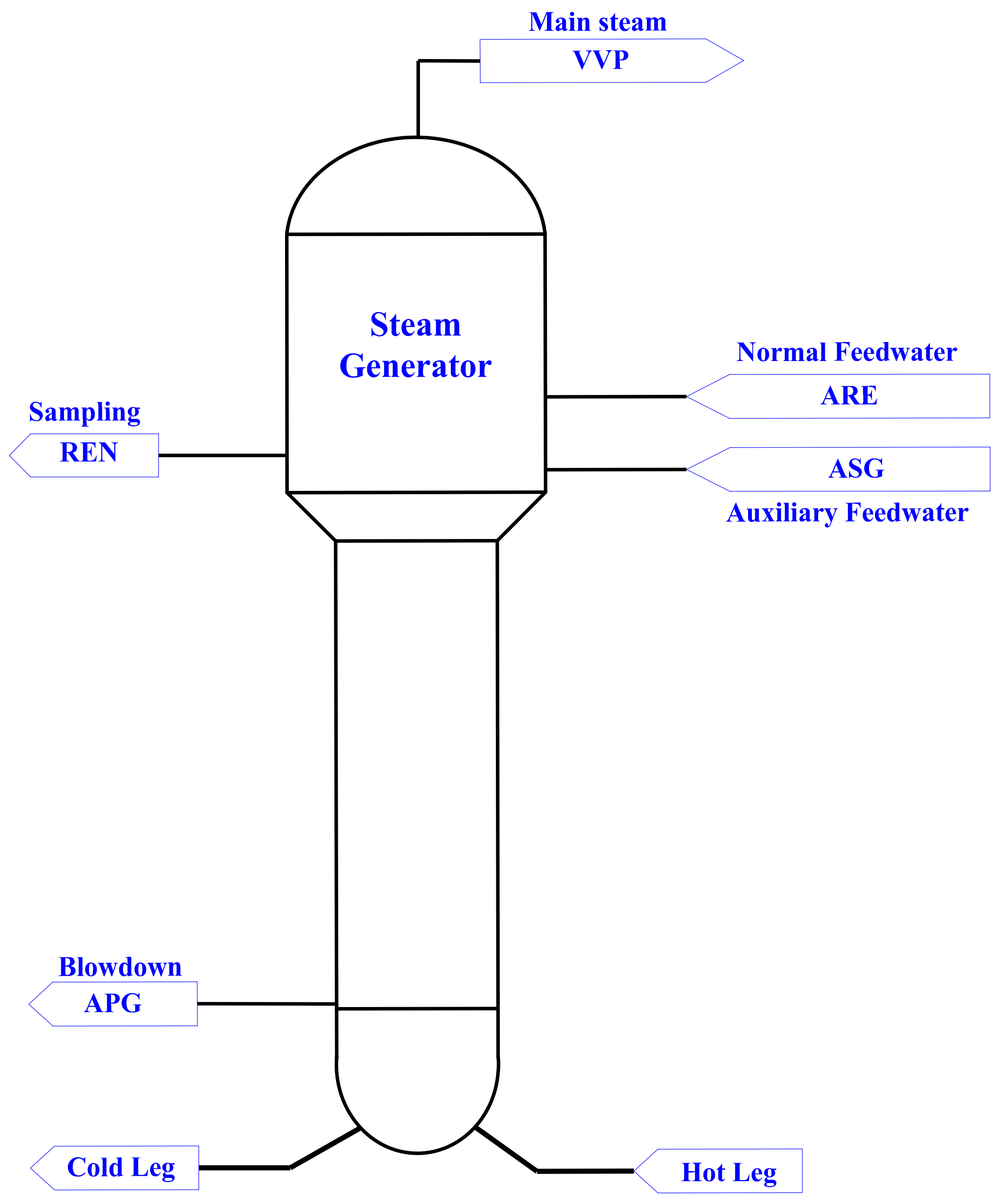
Generic Design Assessment for UK HPR1000 Pre-Construction Safety Report	
F-6B-2	Simplified RCP [RCS] diagram RPV connection



Generic Design Assessment for UK HPR1000 Pre-Construction Safety Report	
F-6B-3	Simplified RCP [RCS] diagram PZR depressure system



Generic Design Assessment for UK HPR1000 Pre-Construction Safety Report	
F-6B-4	Simplified RCP [RCS] diagram Reactor Coolant Pump and its interface



Generic Design Assessment for UK HPR1000 Pre-Construction Safety Report	
F-6B-5	Simplified RCP [RCS] diagram Steam Generator and its interface

Appendix 6C Tables

T-6C-1 Interfaces between Chapter 6 and Other Chapters

PCSR Chapter	Interface
Chapter 1 Introduction	Chapter 1 provides the information relevant to the GDA scope, high level safety case route map and the methodology for route map development. Based on the methodology, Chapter 6 develops a route map of its own. The result of RCP [RCS] design is used to support the high level claim presented in Chapter 1.
Chapter 2 General Plant Description	Chapter 2 provides a brief introduction for the RCP [RCS]. Chapter 6 provides a further description of the reactor coolant system.
Chapter 4 General Safety and Design Principles	Chapter 4 provides the general safety and design principles including the concept of DiD, safety classification of SSC and engineering substantiation. These principles shall be considered in the Chapter 6 RCP [RCS] design, and applicable issue shall be substantiated.
Chapter 5 Reactor Core	Chapter 5 provides information relevant to the reactor core (including fuel assembly) design result. The design result is considered in Chapter 6 RCP [RCS] system and component design.
Chapter 7 Safety Systems	Chapter 6 provides supporting functional requirements relevant to safety and operation functions for safety systems. Chapter 7 provides the design substantiation relevant to these functions.
Chapter 8 Instrumentation and Control	Chapter 6 provides control function requirements that shall be fulfilled by I&C systems. Chapter 8 provides design substantiation relevant to these control functions.
Chapter 9 Electric Power	Chapter 9 provide the design information relevant to the electrical power systems. General power supply information of the RCP [RCS] is described in Chapter 9. Power supply requirements of the RCP [RCS] are described in Chapter 6.
Chapter 10 Auxiliary Systems	Chapter 6 provides supporting functional requirements relevant to safety and operation functions for interfacing auxiliary system.

PCSR Chapter	Interface
	Chapter 10 provides the design substantiation relevant to these functions.
Chapter 11 Steam and Power Conversion System	Chapter 6 provides supporting functional requirements relevant to safety and operation functions for steam and power conversion system. Chapter 11 provides the design substantiation relevant to these functions.
Chapter 12 Design Basis Condition Analysis	Chapter 12 provides the justification of the current RCP [RCS] design in terms of the Design Basis Condition (DBC) analyses. Chapter 6 provides the substantiation of the RCP [RCS], which is takes into consideration the fault analysis.
Chapter 13 Design Extension Conditions and Severe Accident Analysis	Chapter 13 provides the justification of the current RCP [RCS] design in terms of the Design Extension Condition (DEC) analyses. Chapter 6 provides the substantiation of the RCP [RCS], which is takes into consideration the Design Extension Condition analyses.
Chapter 14 Probabilistic Safety Assessment	Chapter 6 provides the design of the RCP [RCS] for the PSA analysis. Chapter 14 provides the estimate feedback showing whether potential enhancement areas are present or not.
Chapter 15 Human Factors	Chapter 15 provides the principles and methodology of Human Factor Integration that shall be considered in system and component design. Chapter 6 provides the substantiation of the RCP [RCS], which is taken into account for further estimates of the Human Factors.
Chapter 16 Civil Works & Structures	Chapter 16 provides design information relevant to the reactor building. The result of the civil structure design is considered in the Chapter 6 RCP [RCS] system and component design.
Chapter 17 Structural Integrity	The demonstration of main equipment structural integrity is presented in Chapter 17. Chapter 6 provides general component design information of the RCP [RCS] (excluding the structural integrity design information).
Chapter 18 External Hazards	Chapter 18 provides external hazards list of UK HPR1000, relevant design principles, potential risk information, and the ALARP conclusion from the

PCSR Chapter	Interface
	external hazards point of view. Chapter 6 provides the RCP [RCS] design substantiation of applied external hazard protection design principles, which is used for external hazards assessment.
Chapter 19 Internal Hazards	Chapter 19 provides internal hazards list of UK HPR1000, relevant design principles, potential risk information, and the ALARP conclusion from the internal hazards point of view. Chapter 6 provides the RCP [RCS] design substantiation of applied internal hazard protection design principles, which is used for internal hazards assessment.
Chapter 20 MSQA & Safety Case Management	PCSR Chapter 20 presents Safety Case and Design Control Management including relevant requirements, process and coding system of the Requirement Management. Chapter 6 applies the arrangements of Requirement Management set out in Chapter 20[(1)].
Chapter 21 Water Chemistry	Chapter 21 provides the water chemistry specification for the primary coolant. These specification is considered in the material selection of SSC.
Chapter 22 Radiological Protection	Chapter 22 provides radiological protection design considerations relevant to the RCP [RCS]. Chapter 6 provides RCP [RCS] design information used in radiological protection design.
Chapter 23 Radioactive Waste Management	Chapter 23 provides the principle of minimising the radioactive waste generation and the management of reactor coolant effluents as well. Chapter 6 provides the design of RCP [RCS] which contributes to minimise radioactive waste at source and generates reactor coolant effluents.
Chapter 24 Decommissioning	Chapter 24 presents the principles of process design that facilitate decommissioning. Chapter 6 provides the design substantiation of the principles that facilitate decommissioning.
Chapter 25 Conventional Safety and Fire Safety	Chapter 25 provides the conventional health and safety risk management techniques and general prevention principles in the system. Chapter 6 provides the design information to demonstrate the conventional health and safety risk

PCSR Chapter	Interface
	management techniques and general prevention principles are applied in the design process of the system.
Chapter 30 Commissioning	Chapter 30 provides arrangements and requirements for commissioning. This design information shall be considered in Chapter 6.
Chapter 31 Operational Management	Chapter 31 provides the arrangement of operating limits and conditions, EMIT ageing and degradation programme. Chapter 6 provides the reactor coolant system design substantiation relevant to EMIT, ageing and degradation.
Chapter 33 ALARP Evaluation	Chapter 33 provides relevant principles, methodology and the approach for the ALARP demonstration. Chapter 6 provide the ALARP demonstration for the RCP [RCS] based on these principles and the approach.

(1): This Chapter will be supplemented in mechanical engineering schedule with application of the coding system in site licensing phase.

T-6C-2 Applicable Codes and Standards for the UK HPR1000 RCP [RCS]

Codes and Standards	Title
IAEA, No.SSR-2/1 (2016)	Safety of Nuclear Power Plants: Design, IAEA Specific Safety Requirement
IAEA, No. NS-G-1.9 (2004)	Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants
RCC-M, 2007 Edition	Design and Construction Rules for Mechanical Equipment of PWR Nuclear Islands
ASME, 2007 edition and 2008 Addenda, 2017 edition	Boiler and Pressure Vessel Code
RCC-MR, 2007 Edition	Design and Construction Rules for Mechanical Components of Nuclear Installations
RSE-M, 2010 edition and 2012 addendum	In-service Inspection Rules for Mechanical Components of PWR Nuclear Islands
NRC Regulation Guide 1.14, 1975 edition	Reactor Coolant Pump Flywheel Integrity
NUREG 0800, 2010 edition	Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition (Section 5.4.1.1, Pump Flywheel Integrity (PWR)).

T-6C-3 Basic Parameters of the RCP [RCS]

No.	Parameter name	Unit	Values	Remarks
1	Design pressure	MPa(g)	17.13	
2	Design temperature	°C	343	
3	Normal operating pressure	MPa(a)	15.5	
4	RPV inlet coolant temperature	°C	289.5	100%FP, Best Estimate (BE) flowrate
5	RPV outlet coolant temperature	°C	324.5	100%FP, BE flowrate
6	Primary coolant flow rate	m^3/h/loop	25450	BE flowrate

T-6C-4 Classification of Main Equipment

	Function Class	Design Provision Class	Code Design Class	Seismic Category	Structural Integrity Class(1)	Remark
Reactor Pressure Vessel	F-SC1	B-SC1	RCC-M 1	SSE1	HIC(2)	
Reactor Vessel Internals	F-SC1	B-SC1	RCC-M G	SSE1	SIC-1	CS
	/	NC	RCC-M G	/	SIC-1	IS
Control Rod Drive Mechanisms	F-SC1	B-SC1	RCC-M 1	SSE1	SIC-1	
Steam Generator	F-SC1	B-SC1	ASME 1	SSE1	HIC(4)	Primary head and tube sheet
		B-SC2				Secondary side shell
Pressuriser	F-SC1	B-SC1	RCC-M 1	SSE1	HIC	
Reactor Coolant Piping	F-SC1	B-SC1	RCC-M 1	SSE1	HIC	MCL
	F-SC1	B-SC1	RCC-M 1	SSE1	SIC-1	SL
Reactor Coolant Pump	F-SC1	B-SC1	ASME 1	SSE1	HIC(5)	
Pressuriser Safety Valve	F-SC1	B-SC1	RCC-M 1(3)	SSE1	SIC-1	
Severe Accident Dedicated Valve	F-SC1	B-SC1	RCC-M 1(3)	SSE1	SIC-1	
Isolation Valves	F-SC1	B-SC1	RCC-M 1(3)	SSE1	SIC-1	Pressure boundary

(1): Only classification is presented in this table. More information is presented in Chapter 17;

(2): High Integrity Component (HIC);

(3): The equipment is designed and supplied by the equipment vendor. The code for design requirement is M1. The code RCC-M is selected. Requirements of using the RCC-M code in equipment design, manufacture, inspection, etc. will be delivered to the equipment vendor in the form of a technical specification.

(4): The detailed scope of the HIC assemblies is introduced in component safety report, i.e. Reference [36].

(5): The pump casing and the flywheel. Detailed information is presented in Reference [38].

T-6C-5 Main Design Parameters of RPV

Parameters	Unit	Values
Operating pressure	MPa (a)	15.5
Design pressure	MPa (g)	17.13
Design temperature	°C	343.0
Design life	Year	60
Number of CRDM nozzles	set	68
Hydraulic test pressure	MPa (a)	24.6
Hydraulic test temperature	°C	≥(Maximum RT_{NDT})+30
Inner diameter of core shell	mm	4340
Thickness of core shell	mm	220
Cladding thickness	mm	7
Main material	---	16MND5
Cladding material	---	309L+308L
Stud material	---	40 NCDV7-03

T-6C-6 Main Design Parameters of RVI

Parameters	Unit	Value
Design Life	year	60
Design Pressure	MPa (g)	17.13
Design Temperature	°C	343
Operating Pressure	MPa (a)	15.5
Height of Fuel Assembly	feet	12
Bypass (Thermal Hydraulic (TH) flowrate)	%	6.5
Number of CRGA	-	68
Number of ICIA	-	46
Inner diameter of Core Barrel (CB)	mm	3630

T-6C-7 Main Design Parameters of CRDM

Parameters	Unit	Value
Design pressure	MPa (g)	17.13
Design temperature	°C	343
Step length	mm	15.875
Travel length	steps	228
Lifting load capacity	N	1618
Stepping speed	mm/min(step/min)	1143 (72)
Trip delay time	ms	≤150
Design life of pressure housing assembly	year	60
Cumulative stepping number		6 million

T-6C-8 Withdrawing Sequence of CRDM

Coil activation sequence		**CRDM movement**
Step 1	{ }	{ }
Step 2	{ }	{ }
Step 3	{ }	{ }
Step 4	{ }	{ }
Step 5	{ }	{ }
Step 6	{ }	{ }
Repeat Step 1 to 6		{ }

T-6C-9 Inserting Sequence of CRDM

Coil activation sequence		CRDM movement
Step 1	{ }	{ }
Step 2	{ }	{ }
Step 3	{ }	{ }
Step 4	{ }	{ }
Step 5	{ }	{ }
Step 6	{ }	{ }
Repeat step 1 to 6		{ }

T-6C-10 Main Design Parameters of the Steam Generator

Parameters	Unit	Value
Number of SG	/	3
Operating pressure (tube inner side)	MPa (a)	15.5
Static steam pressure downstream the flow limiter: - Beginning Of Life (BOL), BE Flow - End Of Life (EOL), TH Flow	MPa (a)	 ≥ 6.75 ≥ 6.6
Moisture content at SG outlet (before the flow limiter)	%	≤0.1% by weight
Feedwater temperature	°C	228
Design pressure (primary side)	MPa (g)	17.13
Design pressure (secondary side)	MPa (g)	8.9
Design temperature (primary side)	°C	343.0
Design temperature (secondary side)	°C	303.0
Primary side hydrostatic test: - Primary side pressure - Secondary side pressure	 MPa(a) MPa(a)	 24.64 0
Secondary side hydraulic test: - Primary side pressure - Secondary side pressure	 MPa (a) MPa (a)	 0 12.87
Test temperature	°C	≥RT_{NDT} + 30°C
Steam drum maximum outer diameter (nozzles excluded)	mm	4870
Overall height	m	Approx. 22.6
Tube outer diameter	mm	{ }
Heat transfer area	m^2	{ }
Tubes material	/	SB-163 UNS N06690
Shells/Tubesheet/Head material	/	SA-508 Grade 3 Class 2
Cladding (channel head) material	/	ER308L/309L
Cladding (tubesheet) material	/	ERNiCrFe-7

T-6C-11 Main Design Parameters of the PZR

Parameters	Unit	Values
Operating pressure	MPa (a)	15.5
Design pressure	MPa (g)	17.13
Operating temperature	°C	345
Design temperature	°C	360
Design life	Year	60
Total volume	m^3	67
Number of heaters	/	108
Total capacity of heaters (not include spare heaters)	kW	2376
Main material	/	18MND5
Cladding material	/	309L+308L

T-6C-12 Main Design Parameters of MCLs and SL

Parameters	Unit	Values
Design life	Year	60
Main Coolant Lines		
Design pressure	MPa (g)	17.13
Design temperature	°C	343
Inner Diameter	mm	760
Pipe Thickness	mm	72
Main material	/	X2CrNi19.10 (Controlled Nitrogen Content, RCC-M M3321)
Nozzle material	/	Z2CN19.10 (Controlled Nitrogen Content, RCC-M M3301)
Surge Line		
Design pressure	MPa (g)	17.13
Design temperature	°C	360
Inner Diameter	mm	284
Pipe Thickness	mm	36
Material	/	X2CrNiMo18.12 (Controlled Nitrogen Content, RCC-M M3321)

T-6C-13 Main Design Parameters of the Reactor Coolant Pump

Parameters	Unit	Value
Design service life	year	60
Design pressure	MPa (g)	17.13
Design temperature	°C	343
Operating pressure	MPa (a)	15.5
Rated flow	m^3/h	25450
Rated head	m	87.4

T-6C-14 List of the Reactor Coolant Pump Parts' Service Life

No.	Part	Service life	Unit
1	Casing	60	years
2	Seal housing	60	years
3	Casing studs and nuts	60	years
4	Seal housing studs and nuts	60	years
5	Impeller	60	years
6	Diffuser	60	years
7	Suction adapter	60	years
8	Pump shaft	60	years
9	Coupling	60	years
10	Guide bearing	≥12	years
11	Shaft seal system	≥6	years
12	Seal O-rings	≥6	years
13	Casing gasket	≥10	years
14	Motor support stand	60	years

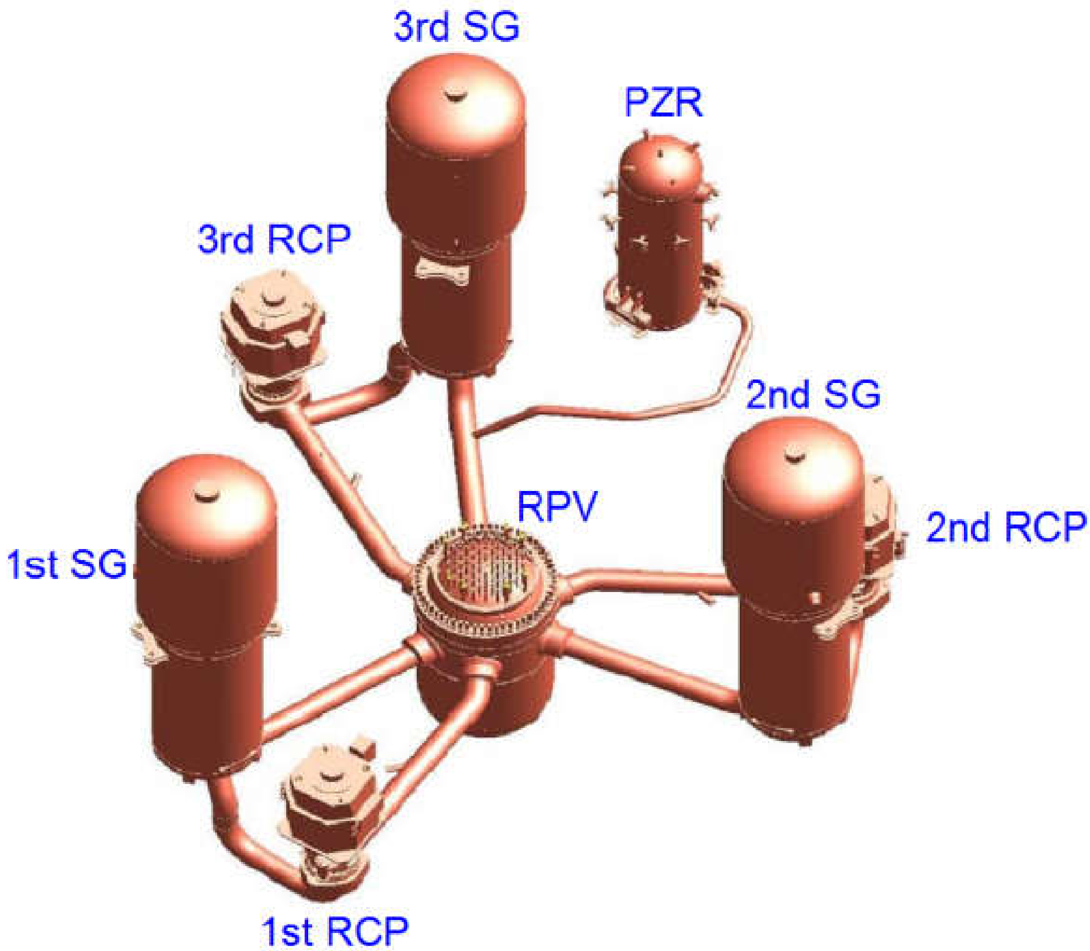
T-6C-15 Main Design Parameters of the PSVs

Parameter	Unit	Value	Remark
Number of valves	/	3	
Design pressure	MPa (g)	17.13	
Design temperature	°C	360	
Nominal flowrate	t/h	210	Saturate steam flowrate at 17.23 MPa (a)
Opening time (without dead time)	s	≤0.1	In hot state, excluding dead time
Closing time (without dead time)	s	≤1.0	In hot state, excluding dead time
Dead time	s	≤0.5	Valve open dead time
	s	≤5.0	Valve close dead time
Set pressure	MPa (a)	17.1	1st PSV
	MPa (a)	17.4	2nd PSV
	MPa (a)	17.7	3rd PSV

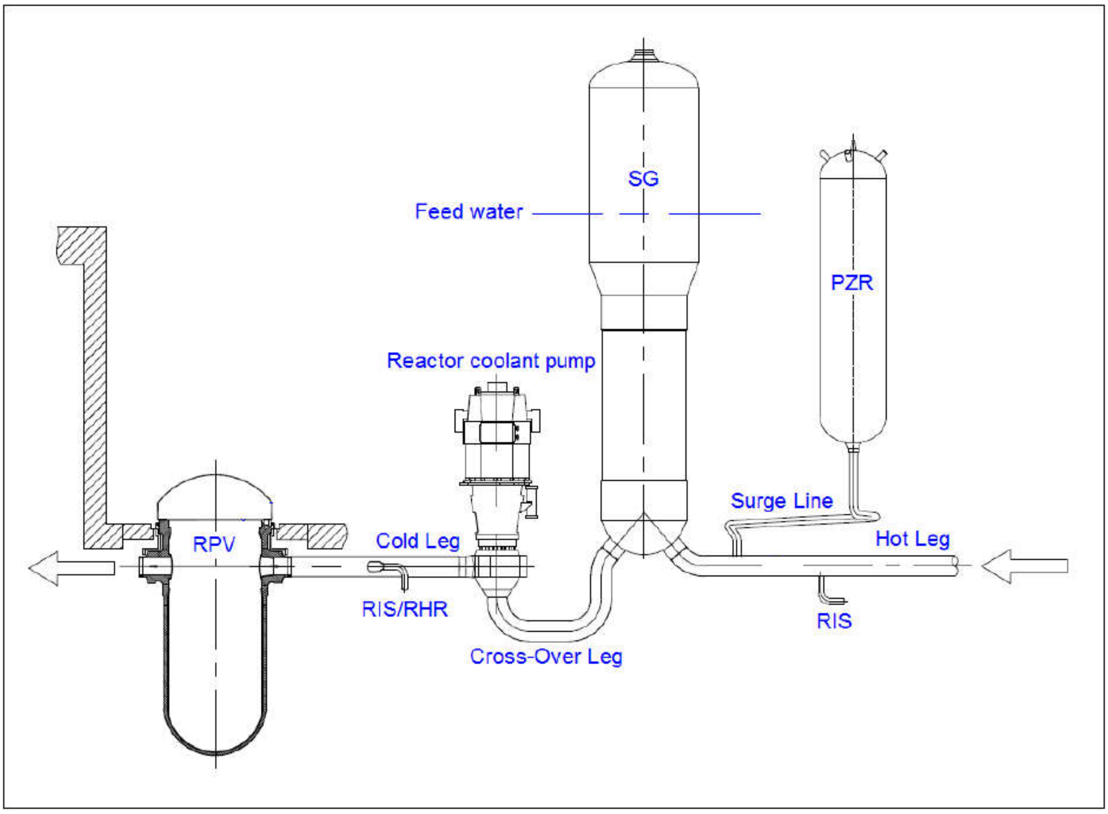
T-6C-16 Main Design Parameters of the SADVs

Parameter	Unit	Value	Remark
Design pressure	MPa (g)	17.13	
Design temperature	°C	360	
Rated discharge flowrate	t/h	630	Saturate steam flowrate at 17.23 MPa (a)
Normal operating temperature	°C	60	Remain closed
Maximum operating temperature	°C	600	Opening, at inlet of the valve.
Opening time	s	≤60	
Closing time	s	≤60	

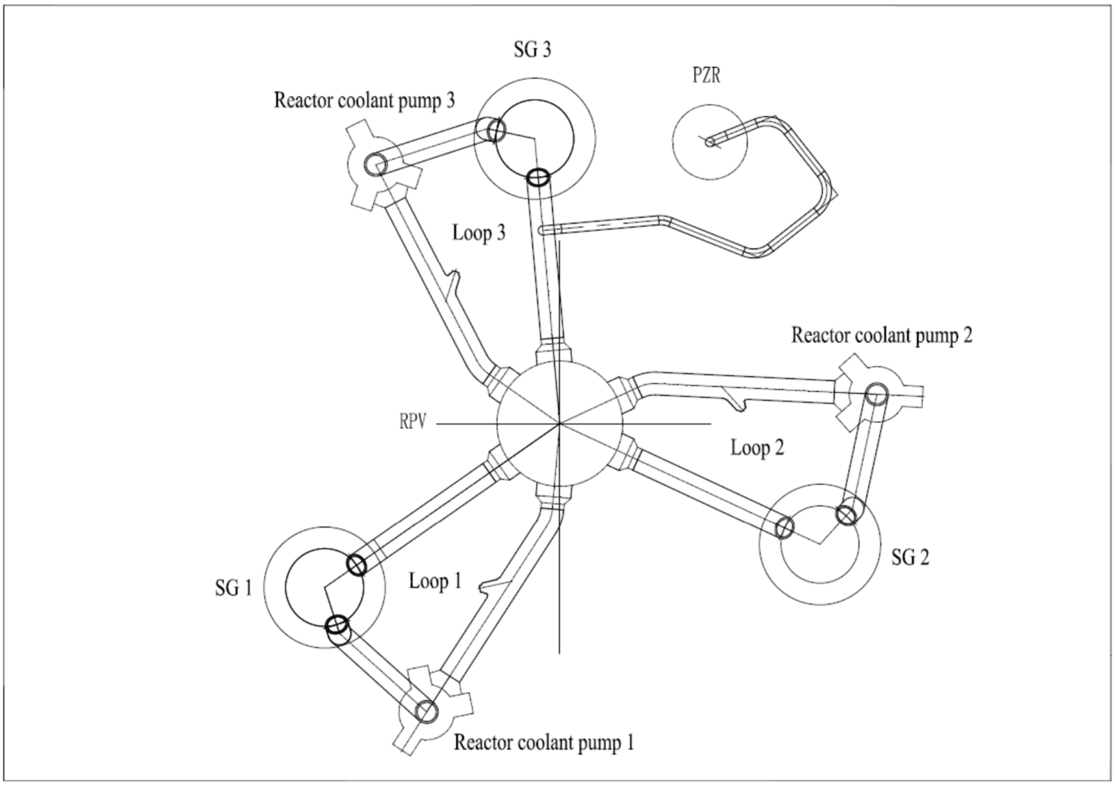
Appendix 6D Figures



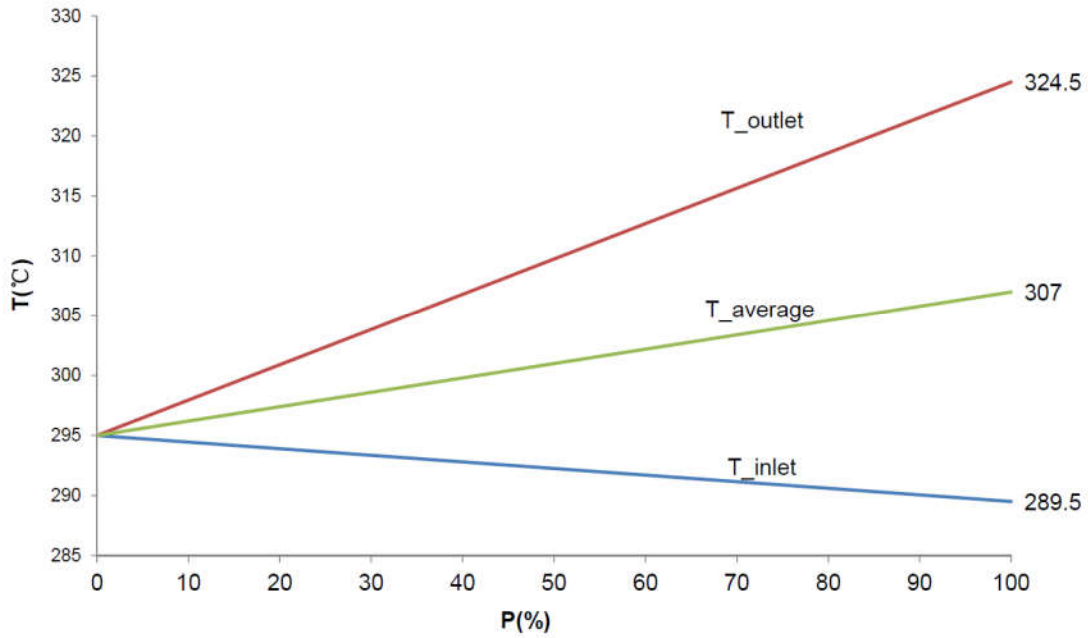
F-6D-1 Overall Schematic of the RCP [RCS]



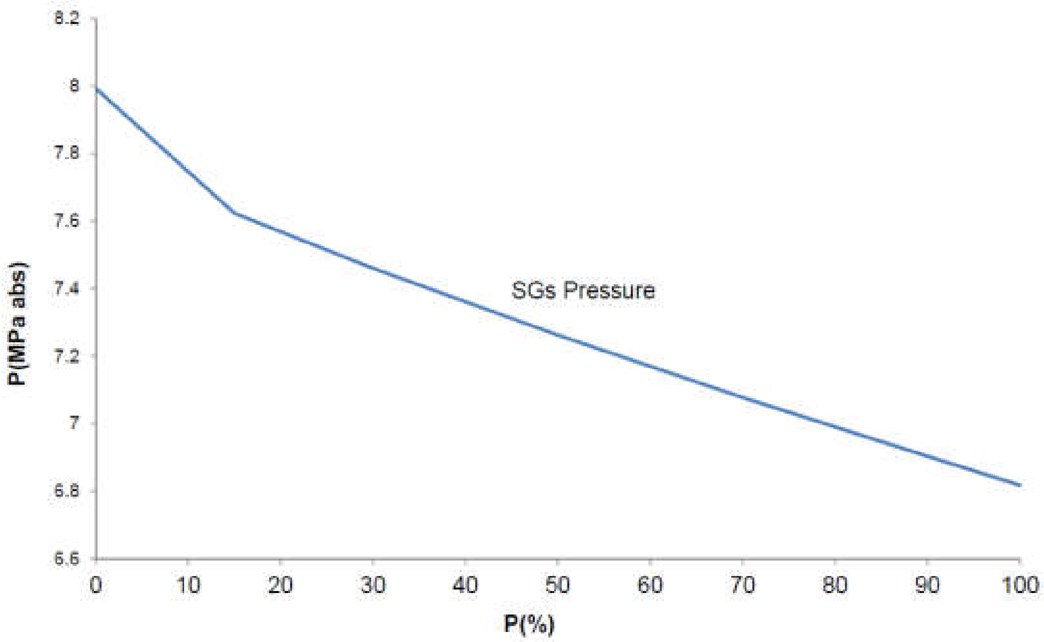
F-6D-2 General Layout Information of the RCP [RCS]



F-6D-3 General Arrangement of the RCP [RCS]

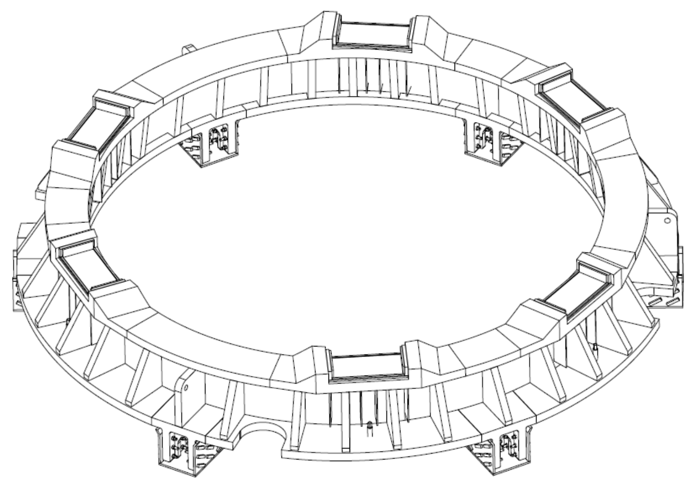


F-6D-4 Primary temperature based on the power at BE flowrate

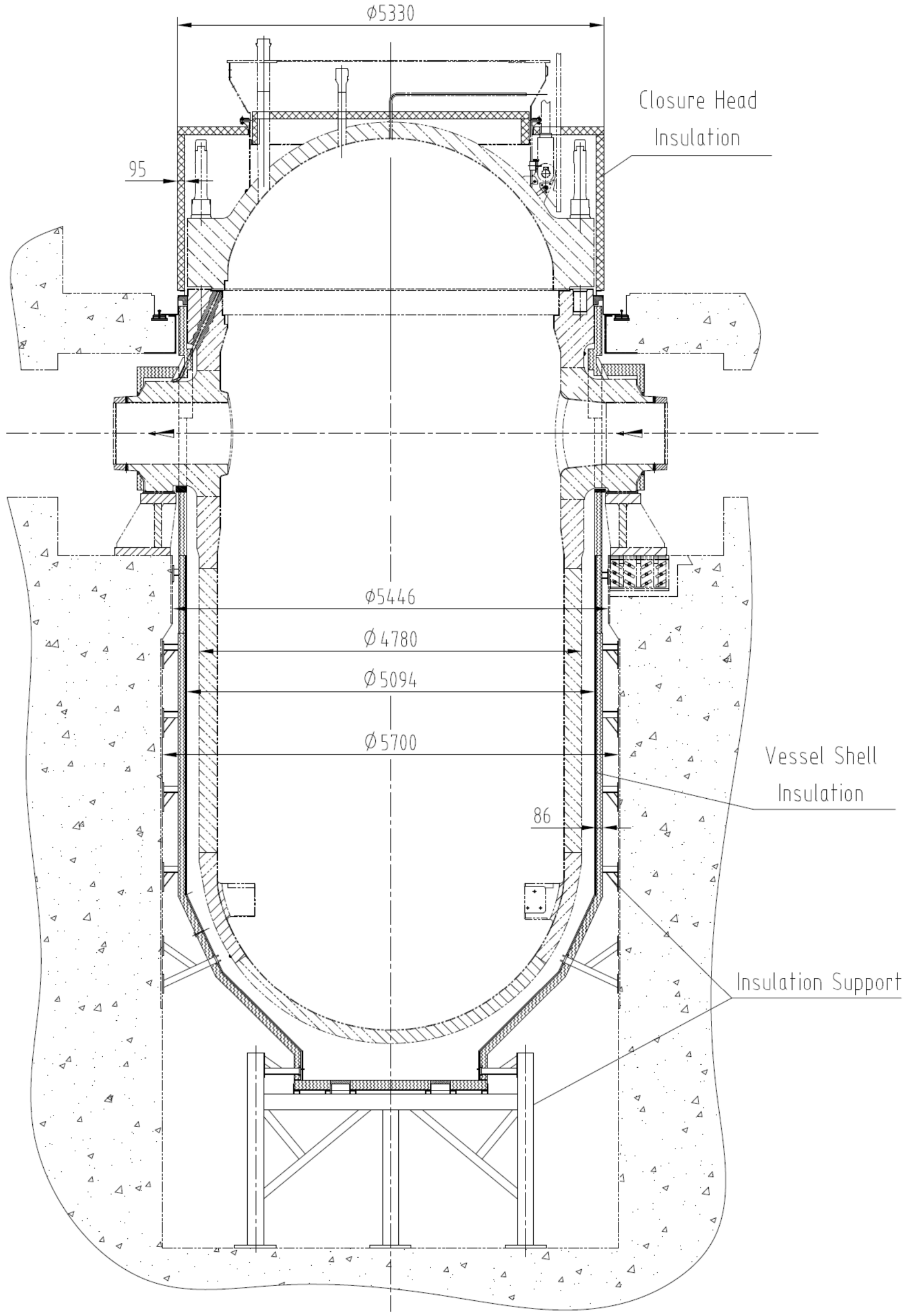


F-6D-5 Steam Generator operating pressure based on the power

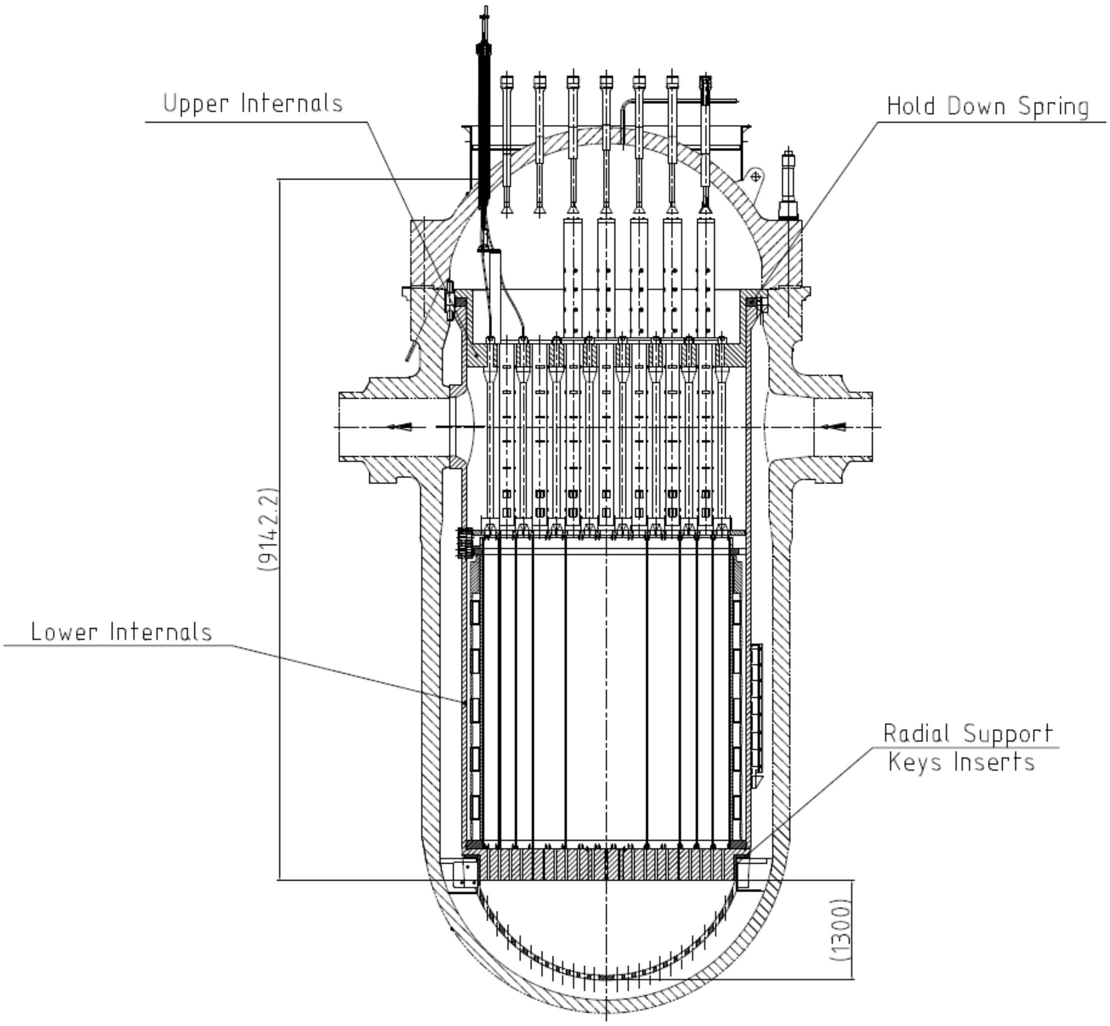
F-6D-6 Structure Schematic Drawing of the RPV



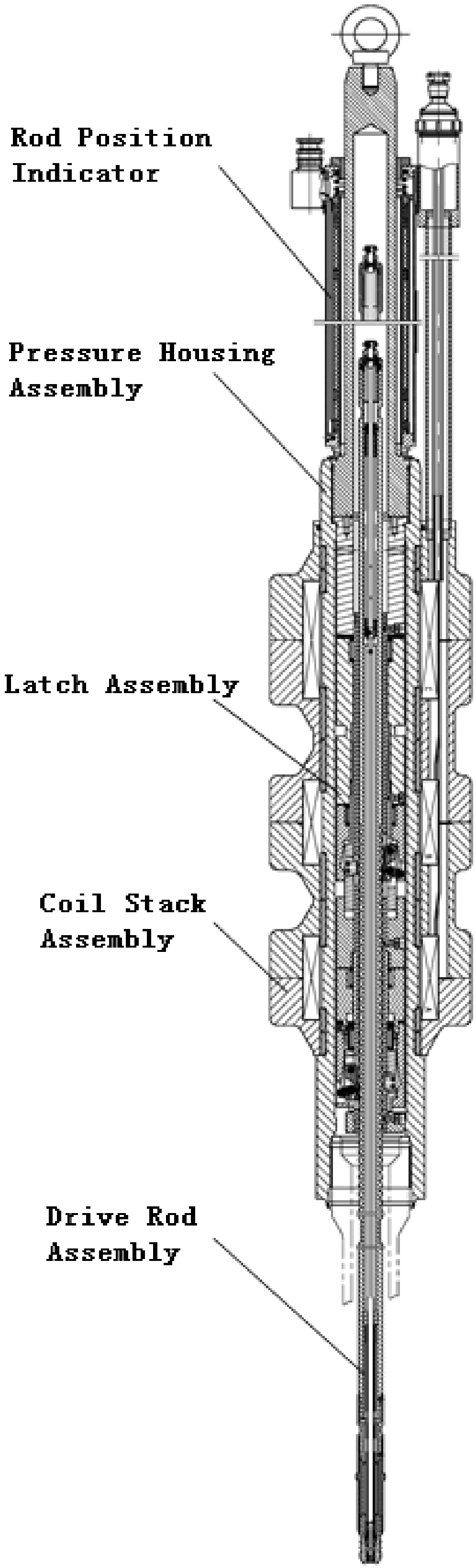
F-6D-7 Structure Schematic Drawing of the RPV Support



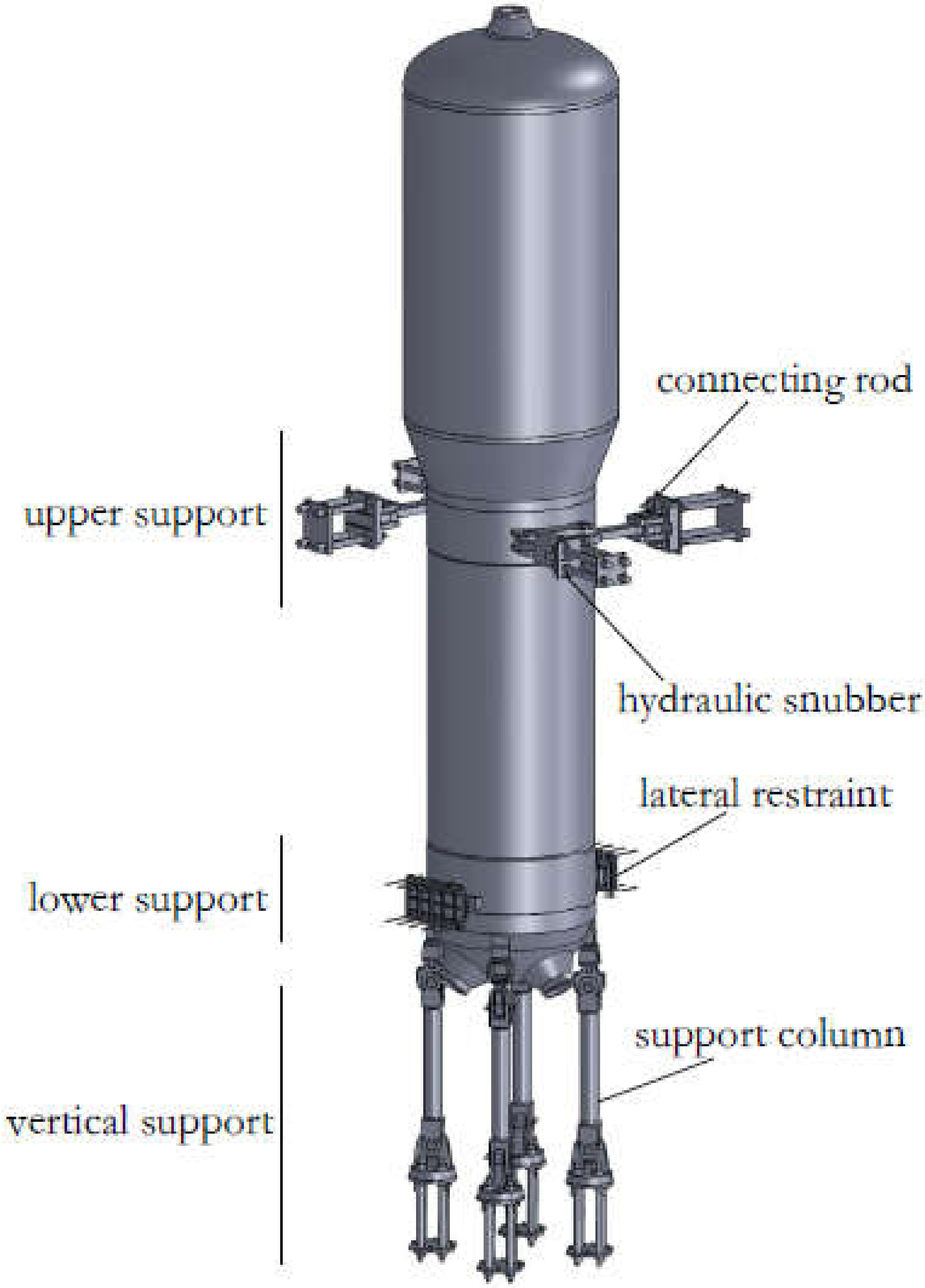
F-6D-8 Structure Drawing of the RPV Insulation



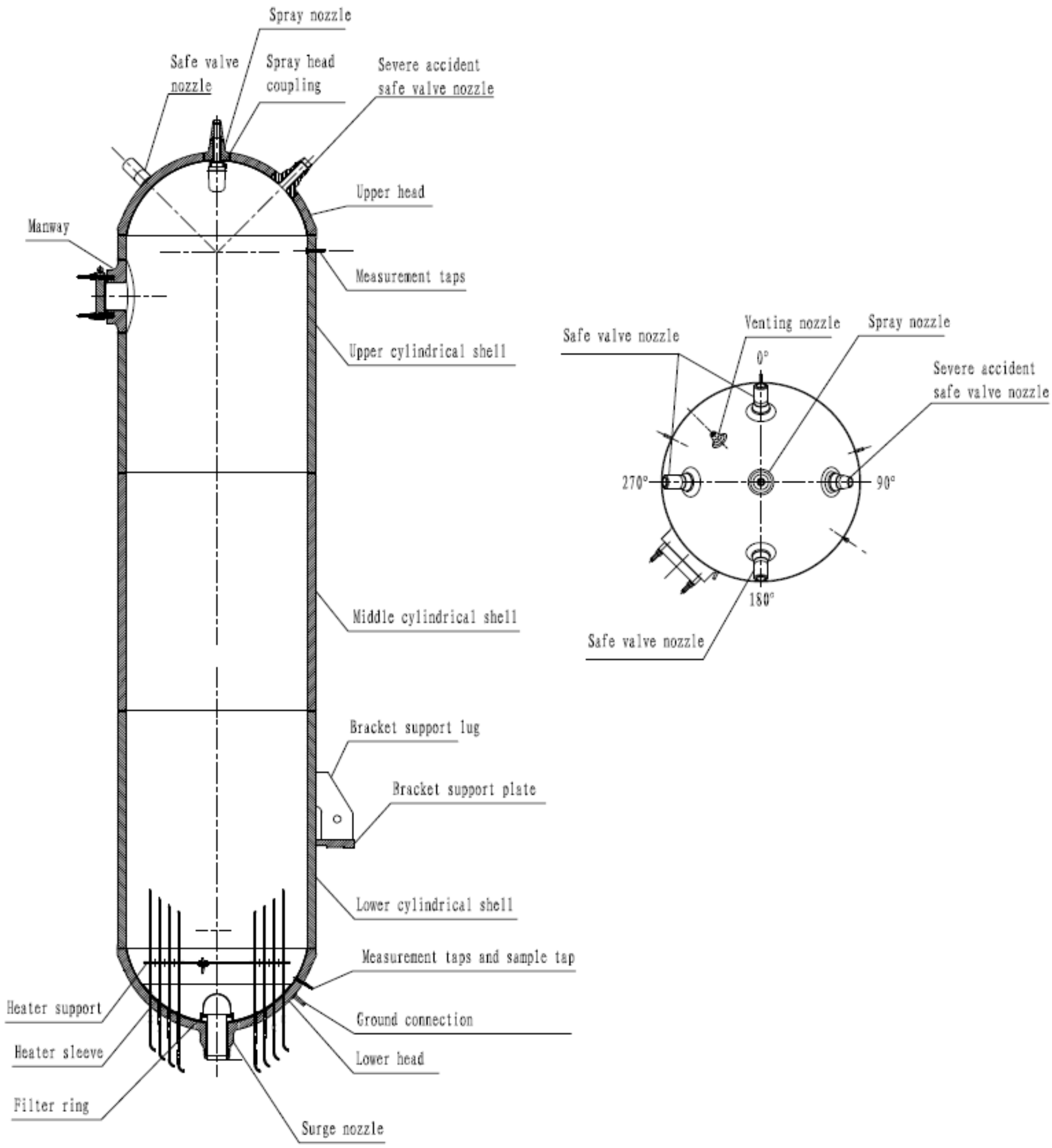
F-6D-9 Structure Schematic Drawing of the RVI



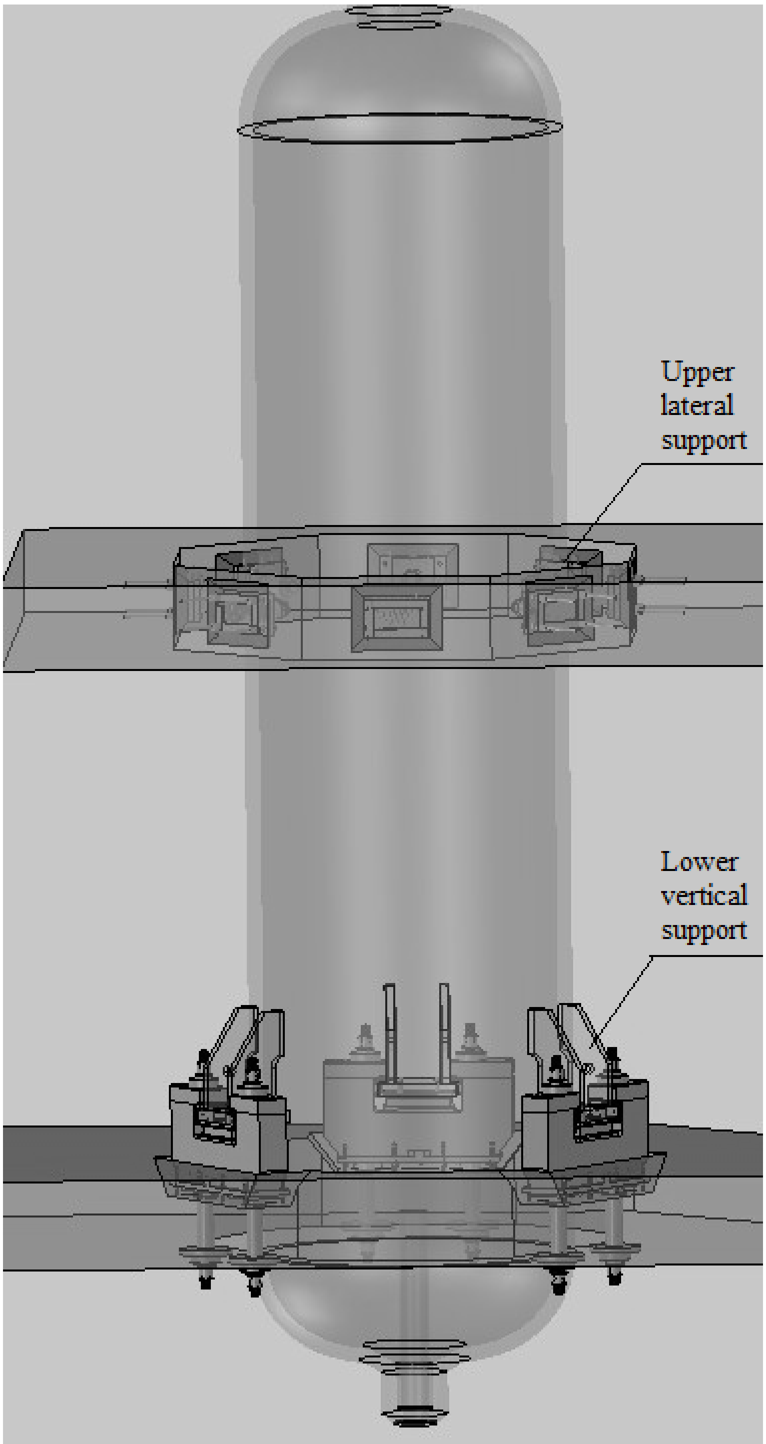
F-6D-10 CRDM Structure Schematic Drawing



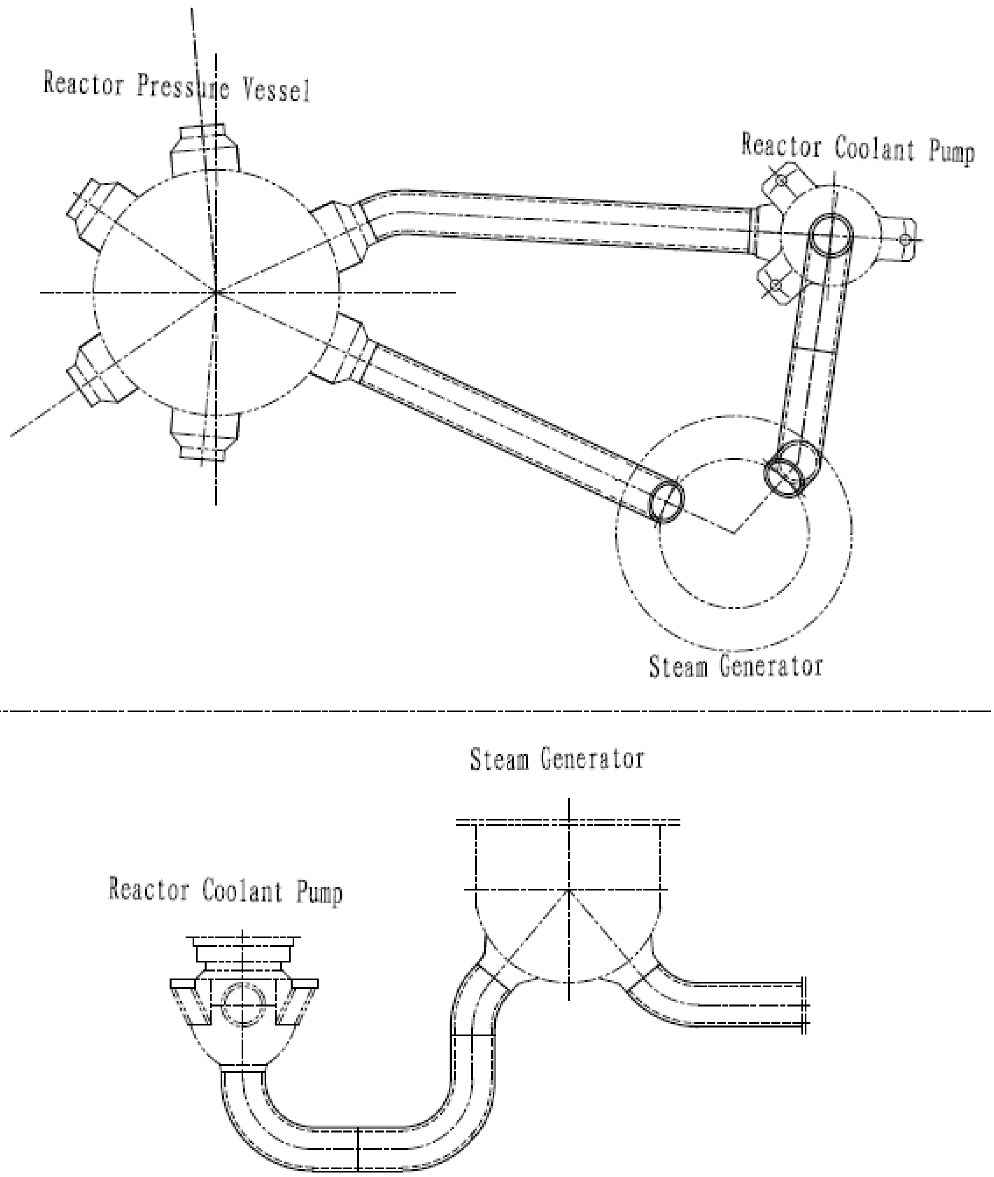
F-6D-11 Steam Generator Support



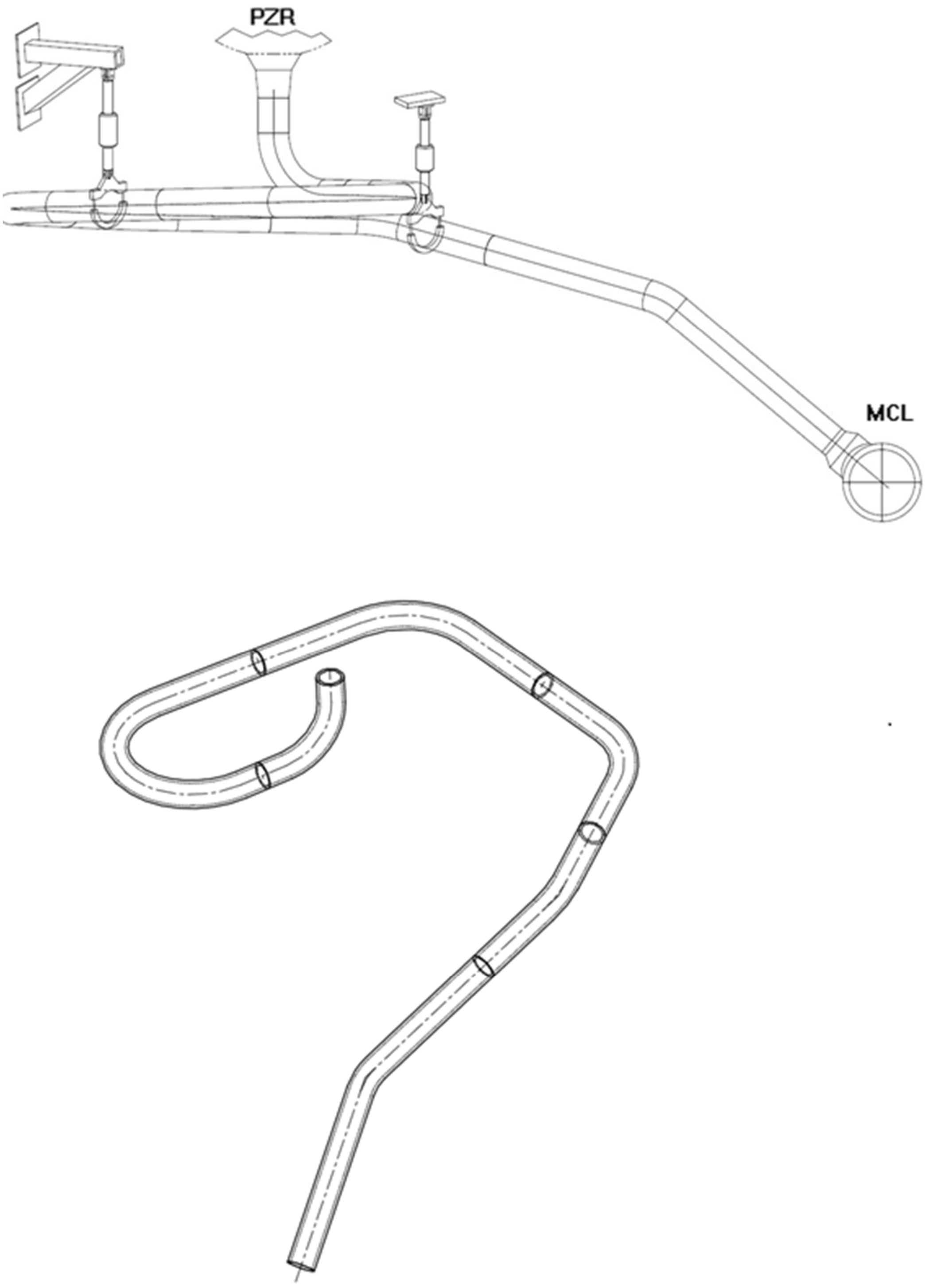
F-6D-12 Structure Schematic Drawing of the Pressuriser



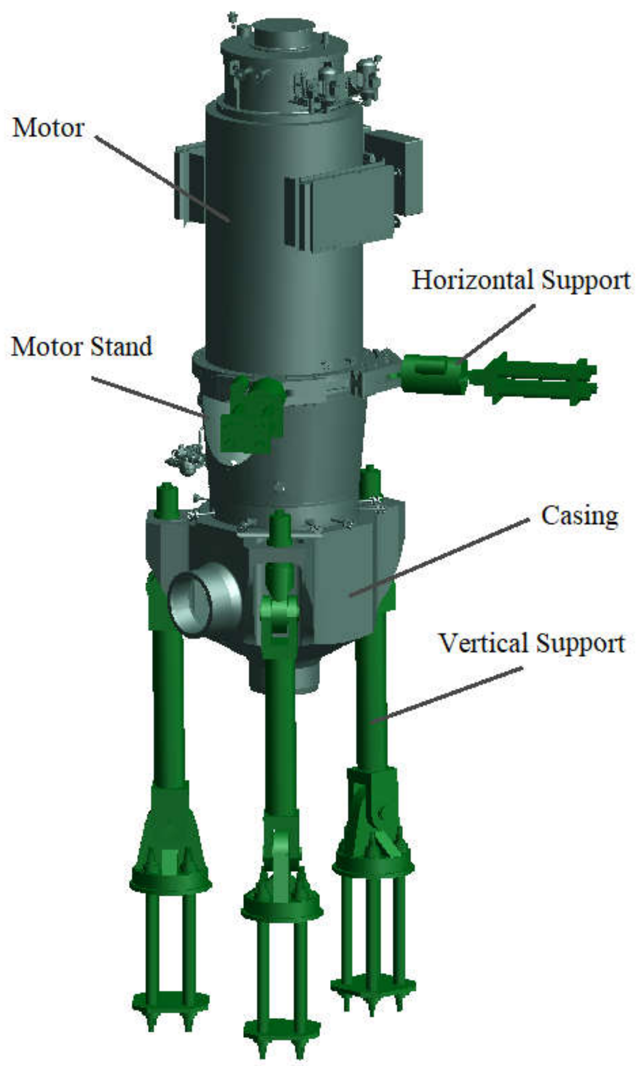
F-6D-13 Structure Schematic Drawing of the Pressuriser Support



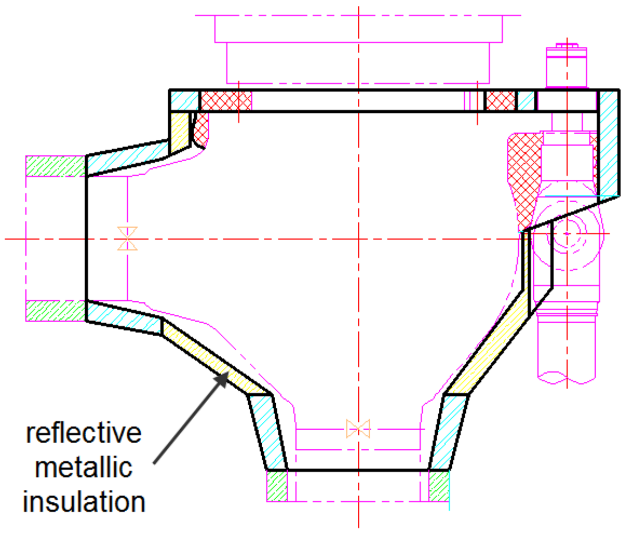
F-6D-14 Schematic Drawing of the Main Coolant Lines



F-6D-15 Structure Schematic Drawing of the Surge Line

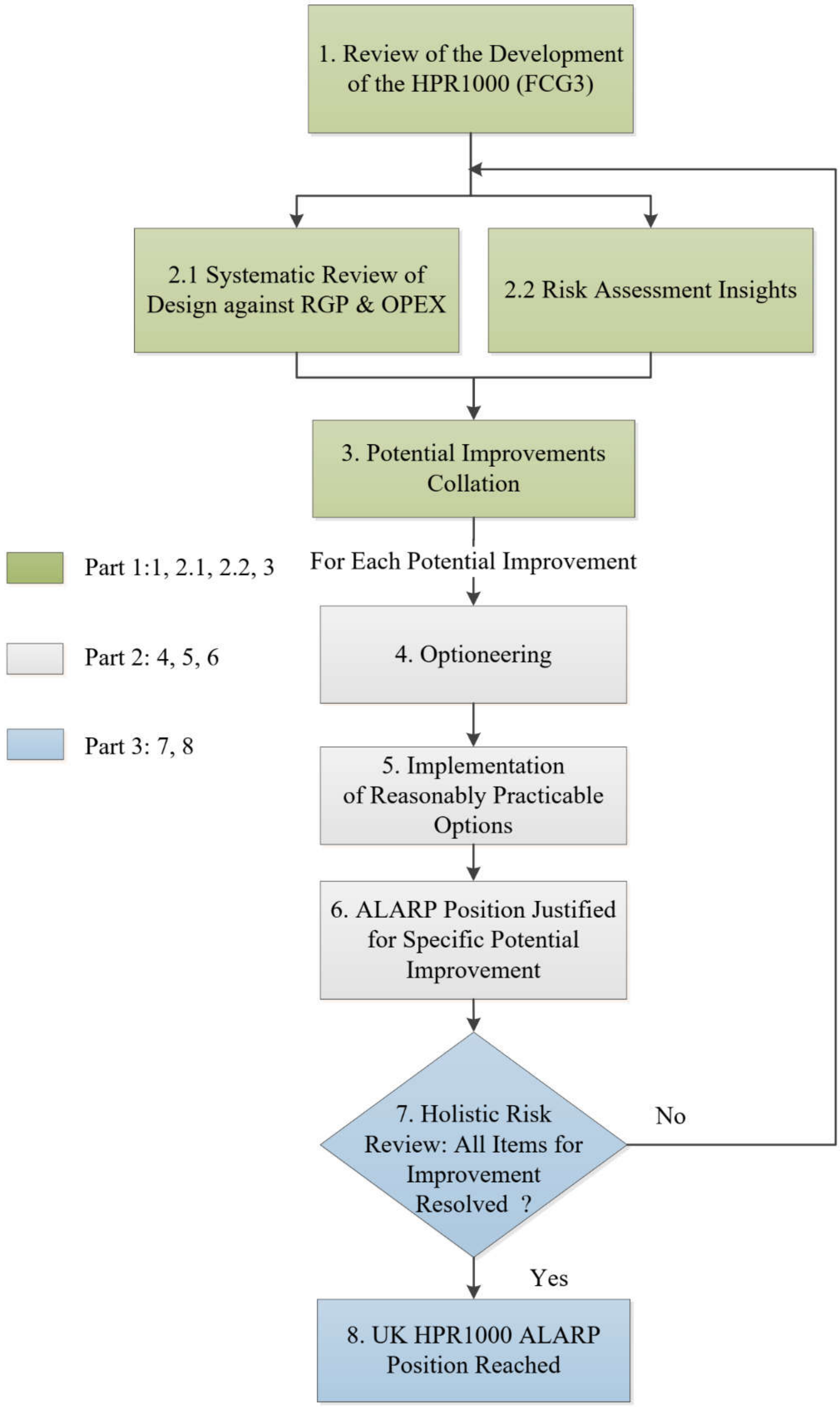


F-6D-16 Outline Drawing of the Reactor Coolant Pump with Supports



F-6D-17 Structure of the Reactor Coolant Pump Insulation

F-6D-18 General Arrangement of the PSVs and SADVs



F-6D-19 Overview of the UK HPR1000 ALARP Approach