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# General Nuclear System Ltd.

UK HPR1000 GDA Project

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Title:

# Pre-Construction Safety Report
# Chapter 2
# General Plant Description

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## 2.1 List of Abbreviations and Acronyms

| | |
|---|---|
| AC | Alternating Current |
| ACPR1000 | Advanced Chinese Pressurised Reactor |
| ADG | Feedwater Deaerating Tank and Gas Stripper System [FDTGSS] |
| APG | Steam Generator Blowdown System [SGBS] |
| ARE | Main Feedwater Flow Control System [MFFCS] |
| ASG | Emergency Feedwater System [EFWS] |
| ASP | Secondary Passive Heat Removal System [SPHRS] |
| ATE | Condensate Polishing System [CPS] |
| BDA | Emergency Diesel Generator Building A |
| BDB | Emergency Diesel Generator Building B |
| BDC | Emergency Diesel Generator Building C |
| BDU | SBO Diesel Generator Building for Train A |
| BDV | SBO Diesel Generator Building for Train B |
| BEJ | Extra Cooling System and Fire-fighting Water Production System Building |
| BEX | Equipment Access Building |
| BFX | Fuel Building |
| BJX | Standby Transformer Platform |
| BLX | Conventional Island Electrical Building |
| BMX | Turbine Generator Building |
| BNX | Nuclear Auxiliary Building |
| BOP | Balance of Plant |
| BPA | Essential Service Water Pumps Station-A |
| BPB | Essential Service Water Pumps Station-B |
| BPW | Circulating Water Pumps Station |
| BPX | Personnel Access Building |
| BRX | Reactor Building |

| | |
|---|---|
| BSA | Safeguard Building A |
| BSB | Safeguard Building B |
| BSC | Safeguard Building C |
| BTA | Main Transformer Platform |
| BTX | Backup Transformer Platform |
| BWX | Radioactive Waste Treatment Building |
| CGN | China General Nuclear Power Corporation |
| CI | Conventional Island |
| CPR1000 | Chinese Pressurised Reactor |
| $CPR1000^{+}$ | Chinese Improved Pressurised Reactor |
| CRF | Circulating Water System [CWS] |
| DBC | Design Basis Condition |
| DBE | Design Basis Earthquake |
| DC | Direct Current |
| DCL | Main Control Room Air Conditioning System [MCRACS] |
| DEC | Design Extension Condition |
| DEC-A | Design Extension Condition A |
| DEC-B | Design Extension Condition B |
| DEL | Safety Chilled Water System [SCWS] |
| DER | Operational Chilled Water System [OCWS] |
| DVD | Diesel Building Ventilation System [DBVS] |
| DVL | Electrical Division of Safeguard Building Ventilation System [EDSBVS] |
| DWK | Fuel Building Ventilation System [FBVS] |
| DWL | Safeguard Building Controlled Area Ventilation System [SBCAVS] |
| DWN | Nuclear Auxiliary Building Ventilation System [NABVS] |
| DWQ | Waste Treatment Building Ventilation System [WTBVS] |

| | |
|---|---|
| DWW | Access Building Controlled Area Ventilation System [ABCAVS] |
| DXE | Extra Cooling Water and NI Firefighting Building Ventilation System [ECW&FFB VS] |
| DXS | Essential Service Water Pumping Station Ventilation System [ESWVS] |
| EBA | Containment Sweeping and Blowndown Ventilation System [CSBVS] |
| ECS | Extra Cooling System [ECS] |
| EDE | Annulus Ventilation System [AVS] |
| EDG | Emergency Diesel Generator |
| EHR | Containment Heat Removal System [CHRS] |
| EUF | Containment Filtration and Exhaust System [CFES] |
| EUH | Containment Combustible Gas Control System [CCGCS] |
| EVF | Containment Internal Filtration System [CIFS] |
| EVR | Containment Cooling and Ventilation System [CCVS] |
| FCG | Fangchenggang Nuclear Power Plant |
| FCG3 | Fangchenggang Nuclear Power Plant Unit 3 |
| FLCV | Full Load Control Valve |
| FLIV | Full Load Isolation Valve |
| GCT | Turbine Bypass System [TBS] |
| GDA | Generic Design Assessment |
| HPR1000 | Hua-long Pressurised Reactor |
| HPR1000 (FCG3) | Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 |
| HVAC | Heating, Ventilation and Air Conditioning |
| HYH | Hongyanhe Nuclear Power Plant |
| I&C | Instrumentation and Control |
| IRWST | In-containment Refuelling Water Storage Tank |
| JAC | Fire-fighting Water Production System [FWPS] |

| | |
|---|---|
| JPI | Fire-fighting Water System for Nuclear Island [FWSNI] |
| JPS | Mobile and Portable Fire Extinguishing Equipment [MPFE] |
| JPV | Fire Extinguishing System for Nuclear Island Diesel Generator Building [FSDB] |
| KCC | Nuclear Accident Emergency Management System [NAEMS] |
| KDA | Severe Accident I&C System [SA I&C] |
| KDS | Diverse Actuation System [DAS] |
| KRT | Plant Radiation Monitoring System [PRMS] |
| LA | Ling'ao Nuclear Power Plant |
| LHSI | Low Head Safety Injection |
| LLCV | Low Load Control Valve |
| LLIV | Low Load Isolation Valve |
| LOOP | Loss of Offsite Power |
| MCR | Main Control Room |
| MHSI | Medium Head Safety Injection |
| MSIV | Main Steam Isolation Valve |
| MSRCV | Main Steam Relief Control Valve |
| MSRIV | Main Steam Relief Isolation Valve |
| MSSV | Main Steam Safety Valve |
| ND | Ningde Nuclear Power Plant |
| NI | Nuclear Island |
| NPP | Nuclear Power Plant |
| PAR | Passive Autocatalytic Recombiner |
| PCSR | Pre-Construction Safety Report |
| PMC | Fuel Handling and Storage System [FHSS] |
| PS | Protection System |
| PSAS | Plant Standard Automation System |
| PTR | Fuel Pool Cooling and Treatment System [FPCTS] |

| | |
|---|---|
| PWR | Pressurised Water Reactor |
| RBS | Emergency Boration System [EBS] |
| RCCA | Rod Cluster Control Assembly |
| RCP | Reactor Coolant System [RCS] |
| 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 [IIS] |
| RIS | Safety Injection System [SIS] |
| RPE | Nuclear Island Vent and Drain System [VDS] |
| RPN | Nuclear Instrumentation System [NIS] |
| RPV | Reactor Pressure Vessel |
| RRI | Component Cooling Water System [CCWS] |
| SAS | Safety Automation System |
| SBD | Radioactive Decontamination System [RDS] |
| SBE | Hot Laundry System [HLS] |
| SBO | Station Black Out |
| SEC | Essential Service Water System [ESWS] |
| SED | NI Dematerialised Water Distribution System [DWDS (NI)] |
| SEL | Conventional Island Liquid Waste Discharge System [LWDS (CI)] |
| SEP | Potable Water System [PWS (NI)] |
| SFIS | Spent Fuel Interim Storage |
| SFP | Spent Fuel Pool |
| SG | Steam Generator |

| | |
| --- | --- |
| SGH | NI Hydrogen Distribution System [HDS] |
| SGN | Nitrogen Distribution System [NDS] |
| SGO | Oxygen Distribution System [ODS] |
| SGTR | Steam Generator Tube Rupture |
| SRE | Sewage Recovery System [SRS] |
| SSC | Structures, Systems and Components |
| TEG | Gaseous Waste Treatment System [GWTS] |
| TEP | Coolant Storage and Treatment System [CSTS] |
| TER | Nuclear Island Liquid Waste Discharge System [NLWDS] |
| TES | Solid Waste Treatment System [SWTS] |
| TEU | Liquid Waste Treatment System [LWTS] |
| TGCS | Turbine Generator Control System |
| TLOCC | Total Loss of Cooling Chain |
| UK HPR1000 | UK version of the Hua-long Pressurised Reactor |
| UPS | Uninterruptible Power Supply |
| VCT | Volume Control Tank |
| VDA | Atmospheric Steam Dump System [ASDS] |
| VLLCV | Very Low Load Control Valve |
| VVP | Main Steam System [MSS] |
| YJ | Yangjiang Nuclear Power Plant |

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

## 2.2 Introduction

The purpose of this chapter is to briefly introduce the design of the UK version of the Hua-long Pressurised Reactor (UK HPR1000). This chapter mainly presents the UK HPR1000 evolution process, the main technical characteristics, the summary of system configuration and the main civil structures.

### 2.2.1 Chapter Route Map

The ***Fundamental Objective*** of the UK HPR1000 is that: *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.*

To underpin this objective, five high level claims (Level 1 claims) and a number of Level 2 claims are developed and presented in Chapter 1. This chapter supports the ***Claim 2.1*** and ***Claim 3.1*** derived from the high level ***Claim 2*** and ***Claim 3*** respectively.

***Claim 2:*** *The UK HPR1000 design will be developed in an evolutionary manner, using robust design process, building on relevant good international practice, to achieve a strong safety and environmental performance.*

***Claim 2.1:*** *The historic development process of the HPR1000 (FCG3) was based on relevant good international practice.*

***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.*

***Claim 3.1:*** *The plant design is sufficiently developed based on reference design.*

### 2.2.2 Chapter Structure

The structure of Chapter 2 is as follows:

a) Sub-chapter 2.1 List of Abbreviations and Acronyms:

This section lists the abbreviations and acronyms that are used in this chapter.

b) Sub-chapter 2.2 Introduction:

This section gives a brief introduction of this chapter.

c) Sub-chapter 2.3 Evolution of the UK HPR1000 Design:

This section describes the evolution process of the UK HPR1000 design.

d) Sub-chapter 2.4 Main Technical Characteristics:

This section summarises the main technical characteristics of the UK HPR1000.

e) Sub-chapter 2.5 Reactor Core:

This section provides a summary of the reactor core, and further information is given in Chapter 5.

f) Sub-chapter 2.6 Reactor Coolant System (RCP [RCS]):

This section provides a summary of the RCP [RCS], and further information is given in Chapter 6.

g) Sub-chapter 2.7 Safety Systems:

This section provides a summary of the safety systems, and further information is given in Chapter 7.

h) Sub-chapter 2.8 Main Auxiliary Systems:

This section provides a summary of the main auxiliary systems, and further information is given in Chapter 10.

i) Sub-chapter 2.9 Steam and Power Conversion Systems:

This section summarises the steam and power conversion systems, and further information is given in Chapter 11.

j) Sub-chapter 2.10 Radioactive Waste Management Systems:

This section summarises the radioactive waste management systems, and further information is given in Chapter 23.

k) Sub-chapter 2.11 Instrumentation and Control:

This section summarises the Instrumentation and Control (I&C) systems, and further information is given in Chapter 8.

l) Sub-chapter 2.12 Electric Power:

This section summarises the electrical power systems, and further information is given Chapter 9.

m) Sub-chapter 2.13 Main Civil Structures:

This section gives introduction to main civil structures and their layouts, and further information is given in Chapter 16.

n) Sub-chapter 2.14 Spent Fuel Interim Storage (SFIS):

This section gives a brief introduction to the spent fuel interim storage, and further information is given in Chapter 29.

o) Sub-chapter 2.15 Emergency Preparedness:

This section gives a brief introduction to the emergency preparedness, and further information is given in Chapter 32.

p) Sub-chapter 2.16 Concluding Remarks:

This section gives concluding remarks for this chapter.

q) Sub-chapter 2.17 References:

This section lists the supporting references of this chapter.

### 2.2.3 Interfaces with Other Chapters

The interfaces with other chapters are listed in T-2.2-1.

T-2.2-1 Interfaces between Chapter 2 and Other Chapters

| PCSR Chapter | Interface |
|---|---|
| Chapter 1 Introduction | Chapter 1 presents the fundamental objective, Level 1 claims and Level 2 claims that are mentioned in Sub-chapter 2.2.1. |
| Chapter 5 Reactor Core | Chapter 5 provides a further description of the reactor core mentioned in Sub-chapter 2.5. |
| Chapter 6 Reactor Coolant System | Chapter 6 provides a further description of the RCP [RCS] mentioned in Sub-chapter 2.6. |
| Chapter 7 Safety Systems | Chapter 7 provides a further description of the safety systems mentioned in Sub-chapter 2.7. |
| Chapter 8 Instrumentation and Control | Chapter 8 provides a further description of the I&C systems mentioned in Sub-chapter 2.11. |
| Chapter 9 Electric Power | Chapter 9 provides a further description of the electrical power systems mentioned in Sub-chapter 2.12. |
| Chapter 10 Auxiliary Systems | Chapter 10 provides a further description of the auxiliary systems mentioned in Sub-chapter 2.8. |
| Chapter 11 Steam and Power Conversion System | Chapter 11 provides a further description of the steam and power conversion systems mentioned in Sub-chapter 2.9. |
| Chapter 16 Civil Works & Structures | Chapter 16 provides a further description of the civil works and structures mentioned in Sub-chapter 2.13. |
| Chapter 18 External Hazards | Chapter 18 presents the protection of external hazards which is mentioned in Sub-chapter 2.13.1. |

| PCSR Chapter | Interface |
|---|---|
| Chapter 19 Internal Hazards | Chapter 19 presents the protection of internal hazards which is mentioned in Sub-chapter 2.13.1. |
| Chapter 20 MSQA and Safety Case Management | The organisational arrangements and quality assurance arrangements set out in Chapter 20 are implemented in the design process and in the production of this chapter. |
| Chapter 23 Radioactive Waste Management | Chapter 23 describes the radioactive waste management systems of the plant mentioned in Sub-chapter 2.10. |
| Chapter 29 Interim Storage of Spent Fuel | Chapter 29 provides information about the interim storage of spent fuels mentioned in Sub-chapter 2.14. |
| Chapter 32 Emergency Preparedness | Chapter 32 provides information about the emergency preparedness mentioned in Sub-chapter 2.15. |

## 2.3 Evolution of the UK HPR1000 Design

The UK HPR1000 takes Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)) as the reference design. The Hua-long Pressurised Reactor (HPR1000) technology has a strong pedigree. It is developed based on experiences of a series of successful Chinese commercial Nuclear Power Plant (NPP) designs, construction, operations and maintenance. The evolution of UK HPR1000 is shown in F-2.3-1.

In the 1980s, China General Nuclear Power Corporation (CGN) imported the M310 nuclear reactor technology from France and built the first large commercial nuclear power plant in China - Daya Bay Nuclear Power Plant. After that, CGN built Ling'ao Nuclear Power Plant (LA) Units 1&2 based on the operation experience feedback from Daya Bay and the modifications implemented in similar units in France.

Since then, continuous improvements and design modifications have been performed. By considering operational experience feedback from similar nuclear power plants and applying new proven technologies and new codes and standards, a series of major modifications were determined and implemented to form the design of Chinese Pressurised Reactor (CPR1000). CPR1000 demonstrated a safer design and has been implemented in a number of units such as LA Units 3&4, Hongyanhe Nuclear Power Plant (HYH) Units 1~4, Ningde Nuclear Power Plant (ND) Units 1~4, Yangjiang Nuclear Power Plant (YJ) Units 1&2, and Fangchenggang Nuclear Power Plant (FCG)

Units 1&2.

In 2009, the design of Chinese Improved Pressurised Reactor (CPR1000$^+$) was established based on the CPR1000 design by implementing a number of major modifications with due consideration of improving the nuclear safety (especially in the aspects of severe accident mitigation), optimising the waste treatment design, improving the reliability and economics and facilitating the operation and maintenance. CPR1000$^+$ has been implemented in YJ Units 3&4.

After the Fukushima nuclear accident, to sufficiently consider the lessons learnt from the Fukushima accident and to meet the new national requirements, CGN proposed a number of major improvements based on the design of CPR1000$^+$. The improvements were identified to address the lessons learnt from the Fukushima accident, the consistency review against the latest nuclear safety requirements and the insights from the full scope probabilistic safety assessment. By implementing these modifications, the design of Advanced Chinese Pressurised Reactor (ACPR1000) was formed which demonstrated higher reliability, safety and the ability to manage beyond design basis conditions similar to the Fukushima accident, and possessed the major safety characteristics of the third-generation nuclear power plants. ACPR1000 has been implemented in YJ Units 5&6 and HYH Units 5&6. Further information about the improvements implemented in ACPR1000 is given in Reference [1], these improvements address the lessons learnt from the Fukushima accident and are also considered in the design of HPR1000.

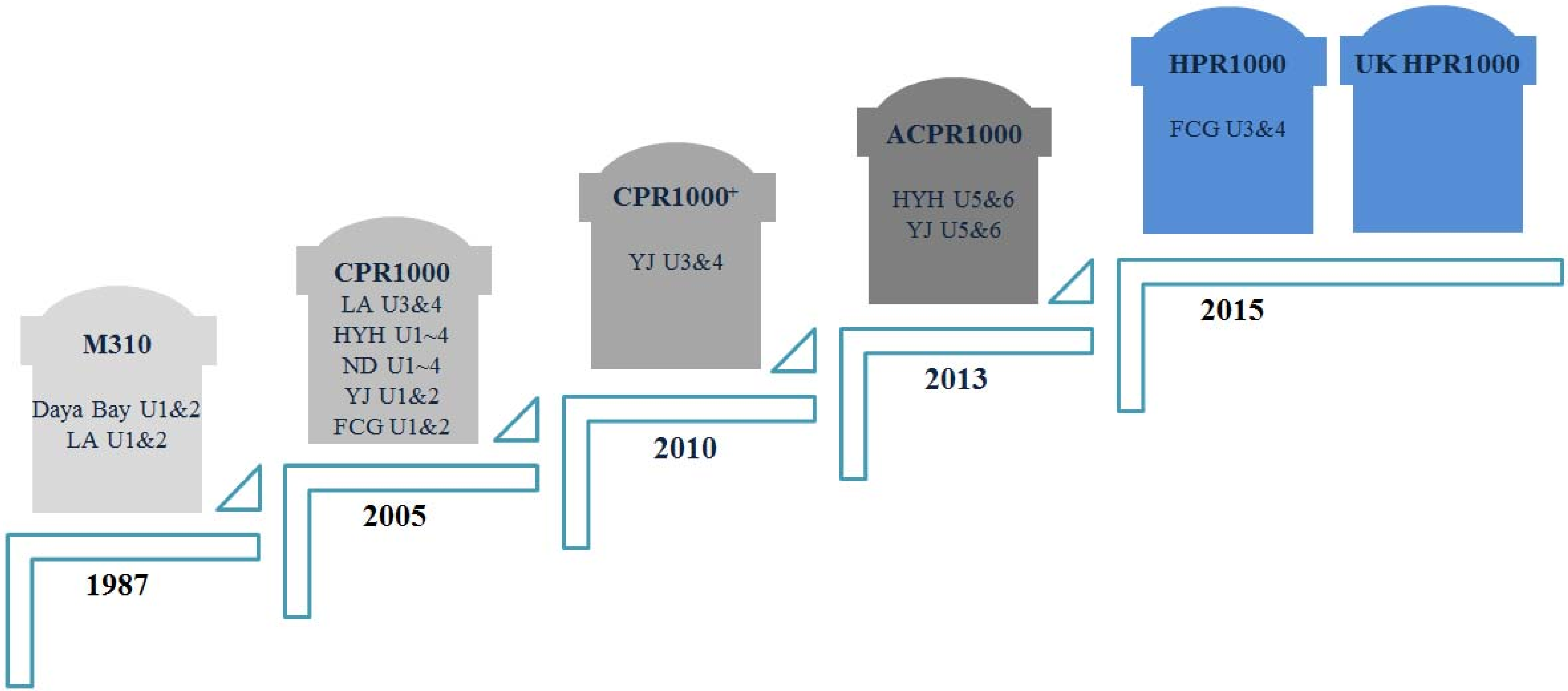
Based on ACPR1000, the third-generation nuclear power plant technology HPR1000 was developed. Compared with the ACPR1000 design, the HPR1000 adopts a single-unit layout and has better site foundation adaptability, regional grid adaptability and relatively low total investment. It improves the incorporation of the concept of defence in depth, and in particular, further strengthens severe accident prevention and mitigation. It makes full use of and integrates advanced design ideas of third-generation nuclear technologies, the experience of the design, construction, commissioning and operation regarding pressurised water reactors in China, and the achievements of nuclear power development and research in recent years. HPR1000 is implemented in FCG Units 3&4.

The historical development process of HPR1000 went through a series of NPP technologies as summarised in T-2.3-1. Further information about the evolution history of HPR1000 including the major improvements during each process are addressed in Reference [1].

T-2.3-1 Evolution of the CGN Reactor Fleet

| **Reactor Type** | **Number of Units** | **Commercial Operation Date of the First Unit** |
|---|---|---|
| M310 | 4 | February 1994 |
| CPR1000 | 14 | September 2010 |
| $CPR1000^{+}$ | 2 | February 2016 |
| ACPR1000 | 4 | July 2018 |
| HPR1000 | 2 | Under construction |

The design of UK HPR1000 takes the HPR1000 (FCG3) as the reference design. In order to satisfy the requirements of UK context, considering Relevant Good Practice (RGP), the potential improvements will be identified and justified. The final improvements will be incorporated into the design of UK HPR1000. Chapter 2 will be updated with the UK HPR1000 design process to reflect the design improvements.




F-2.3-1 Evolution of UK HPR1000

## 2.4 Main Technical Characteristics

The UK HPR1000 is a third-generation 3-loop Pressurised Water Reactor (PWR), which has a design life of 60 years and a nominal electrical power output of 1180 MW (depending on heat sink temperature). The main design characteristics of the UK HPR1000 are as follows:

a) The reactor core of the UK HPR1000 contains 177 fuel assemblies and 68 Rod Cluster Control Assemblies (RCCAs). Each fuel assembly consists of 264 fuel rods, 24 guide tubes and 1 instrument tube, which are arranged in 17×17 square matrix. The $UO_2$ fuel is in the form of pellets, which are enclosed in the fuel rod cladding tube. The core reactivity is controlled by RCCAs and the boron dissolved in the reactor coolant. A further description of the reactor core is given in Sub-chapter 2.5.

b) The RCP [RCS] consists of three loops linked to the Reactor Pressure Vessel (RPV). Each loop is composed of a Steam Generator (SG), a reactor coolant pump, and the main coolant line (a hot leg, a cold leg and a cross leg). The pressuriser is connected to the hot leg of the third loop through the surge line. A further description of the RCP [RCS] is given in Sub-chapter 2.6.

c) The UK HPR1000 has a number of active and passive safety systems which perform safety functions under accident conditions. A further description is given in Sub-chapter 2.7. The safety systems are designed based on international practice, taking into account the lessons learned from the Fukushima accident.

d) The auxiliary systems support the normal operation of the unit. Similar with the majority of advanced Pressurised Water Reactors (PWRs), the auxiliary systems of UK HPR1000 consist of nuclear auxiliary systems, process auxiliary systems, Heating, Ventilation and Air Conditioning (HVAC) systems and fire protection systems. A further description is given in Sub-chapter 2.8.

e) The steam and power conversion system is designed to remove heat from the RCP [RCS] via three SGs and to convert it to electrical power in the turbine-generator set. A further description of steam and power conversion systems is given in Sub-chapter 2.9.

f) The radioactive waste management systems of UK HPR100 are composed of the liquid radioactive waste management system, the gaseous radioactive waste management system and the solid radioactive waste management system. A further description of radioactive waste management systems is given in Sub-chapter 2.10.

g) Sufficient defence in depth is applied in I&C systems and electrical power systems to support the delivery of safety functions. A further description of I&C systems and electrical power systems is given in Sub-chapter 2.11 and

Sub-chapter 2.12.

h) The UK HPR1000 adopts a single-unit layout. In line with international practice, the reactor building features a double-walled containment. The protection against external hazards is considered in the design of buildings on the Nuclear Island (NI). Effects of hazards are also taken into account in the general layout with appropriate separation of buildings. In addition, the reactor building, fuel building and Safeguard Building C (BSC) are reinforced to withstand an aircraft crash. A further description is given in Chapter 16.

i) Various emergency provisions are equipped in the UK HPR1000 design for emergency preparedness purpose. A further description is given in Sub-chapter 2.15.

The main design parameters of the UK HPR1000 are summarised in T-2.4-1.

T-2.4-1 UK HPR1000 Main Technical Parameters

| No. | Parameters | Unit | Values |
|---|---|---|---|
| 1 | Plant Type | --- | 3-loop PWR |
| 2 | Layout | --- | Single Unit |
| 3 | Design Life | year | 60 |
| 4 | Design Basis Earthquake (DBE) | g | 0.3 |
| 5 | Operator Grace Time | min | 30<br>(for operator action from the main control room) |
| 6 | Nominal Power Output | $MW_e$ | 1180<br>(depending on heat sink temperature) |
| 7 | Core Rated Thermal Power | $MW_{th}$ | 3150 |
| 8 | Core Thermal Margin | % | ≥15 |
| 9 | RPV Coolant Average Temperature (100% full power) | °C | 307.0 |

| No. | Parameters | Unit | Values |
|---|---|---|---|
| 10 | Operational Pressure of Primary Side | MPa abs | 15.5 |
| 11 | Best Estimate Flowrate (single loop) | $m^3/h$ | 25450 |
| 12 | Best Estimate Containment Free Volume | $m^3$ | ~73500 |

## 2.5 Reactor Core

The reactor core of the UK HPR1000 contains 177 fuel assemblies and 68 RCCAs. Each fuel assembly consists of 264 fuel rods, 24 guide thimbles, 1 instrumentation tube, 1 top nozzle, 1 bottom nozzle, 6 mixing grids, 1 top end grid, 1 bottom end grid and 3 flow mixing grids. The guide thimbles, instrumentation tube and fuel rods are arranged in 17×17 square matrix.

The grids include structural grids and flow mixing grids. The structural grids restrain the fuel rod by 2 springs and 4 dimples in each cell, while flow mixing grids do not restrain the fuel rod, functioning as improving the thermal-hydraulic performance.

The top and bottom nozzles play an important role as the structural component, positioning the fuel assembly in the reactor core by engaging with the alignment pin on the upper core plate.

The guide thimbles provide channels for insertion of different types of core components whose type depends on the position of the particular fuel assembly in the core.

The instrumentation tube is located in the centre and provides a channel for insertion of an in-core neutron detector, which is used to monitor the fission process during normal operation.

The $UO_2$ fuel is in the form of pellets, which are enclosed in the fuel rod cladding tube. The fuel rods are supported at intervals along the length by the grid assemblies that maintain the lateral spacing between the rods. The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and top/bottom nozzles to accommodate fuel rod expansion.

Assemblies of different enrichments of $^{235}U$ and different numbers of burnable absorber are positioned one next to another to form a checkerboard-like pattern.

Core reactivity is controlled by chemical poison dissolved in the coolant, RCCAs and burnable absorber rods:

a) The Burnable absorber material ($Gd_2O_3$) is blended within $UO_2$ to flatten the power distribution and to reduce the soluble boron concentration particularly at beginning of cycle. During power operation, the burnable absorbers are depleted, thus positive reactivity is introduced and compensate the negative reactivity from the fuel depletion and the accumulation of fission products.

b) The soluble boron as boric acid is used to control relatively slow reactivity changes due to the moderator temperature defect, the xenon or samarium-induced poisoning and the effects of fuel depletion.

c) The RCCAs are used to achieve shutdown state by providing required shutdown margin and to compensate for the fast reactivity changes.

A further description of the reactor core is given in Chapter 5.

## 2.6 Reactor Coolant System (RCP [RCS])

The pressure retaining boundary of the RCP [RCS] pipes and components constitutes part of the reactor coolant pressure boundary. Under normal operation conditions, the RCP [RCS] serves as a closed circuit combined with its interfacing systems, which supports the following objectives:

a) The reactivity of the reactor core can be controlled and adjusted;

b) The heat generated by the core (including residual heat) can be removed;

c) The radioactive material can be confined;

d) The important system parameters can be monitored;

e) The hazard risk shall be prevented.

Under accident conditions, the main tasks of RCP [RCS] are summarised as below:

a) Serving as a route to ensure the borated water can be injected into RPV to control the reactivity;

b) Removing residual heat by natural circulation through the core if loss of primary forced flow;

c) Providing pressure control function via pressuriser spray and heater to support the heat removal function, and providing depressurisation function in order to maintain the integrity of primary loop;

d) Transferring heat from the reactor coolant to the secondary side systems, via Safety Injection System (RIS [SIS]) in Residual Heat Removal (RHR) mode, or via feed and bleed operation under Design Extension Condition A (DEC-A);

e) Providing defence in depth measures which depressurise RCP [RCS] to avoid high-pressure melt ejection under Design Extension Condition B (DEC-B).

The RCP [RCS] consists of three loops linked to the RPV. Each loop is composed of a SG, a reactor coolant pump, and the main coolant line. The main coolant line consists of a hot leg, a cold leg and a cross leg. The pressuriser is connected to the hot leg of the third loop through the surge line.

A brief introduction of the main equipment of RCP [RCS] is given below.

a) Reactor pressure vessel

The Reactor Pressure Vessel (RPV) is a part of the reactor coolant pressure boundary. It contains the reactor core and serves as the second important barrier for protection against core radioactivity. The main structure of the RPV is a cylinder which is located at the centre of the Reactor Building (BRX).

b) Reactor vessel internal

The reactor vessel internals refer to the parts in the RPV except for the fuel assembly and related components, the core measuring instrument and related components and the irradiated sample monitoring pipes. The reactor vessel internals consist of the lower internals, upper internals and interface components.

c) Control rod drive mechanism

The control rod drive mechanism performs the reactivity control function under normal operation conditions and the reactor trip function under accident conditions. It consists of five separate assemblies which are the pressure housing assembly, latch assembly, drive rod assembly, coil stack assembly and rod position indicator assembly.

d) Steam generator

The steam generator is a natural circulation U-tube heat exchanger which serves to transfer the heat carried by the primary coolant to the secondary side. The tubes and tube plates of steam generator are also part of the reactor coolant pressure boundary.

e) Reactor coolant pump

The reactor coolant pump is a single stage vertical shaft seal pump which provides forced circulation flow for the primary loop to remove the heat generated by the core through the coolant. The main components of the reactor coolant pump include the casing, the hydraulic parts, the shaft seal system and the motor.

f) Pressuriser

The pressuriser is a vertical cylindrical vessel equipped with sprays and heaters, which serves to control and keep the primary pressure within the permissible limits. The pressuriser also contributes to compensate for primary coolant volume changes induced by unit power fluctuations. The pressuriser is a part of the reactor coolant pressure boundary.

A further description of the RCP [RCS] is given in Chapter 6.

## 2.7 Safety Systems

A safety system is a system which acts in response to an accident to protect against a radiological consequence. Brief introduction of safety systems of the UK HPR1000 is given below. A further description of these systems is given in Chapter 7.

### 2.7.1 Safety Injection System (RIS [SIS])

The RIS [SIS] provides borated water injection into the RCP [RCS] to compensate for the water inventory loss under Design Basis Condition (DBC)-2/3/4 and DEC-A. In addition, the RIS [SIS] can operate in Residual Heat Removal (RHR) mode, performing residual heat removal functions in the long-term after an accident.

The RIS [SIS] consists of three independent trains (each train corresponds to one RCP [RCS] loop), and each train is located in one safeguard building respectively. Each train of the RIS [SIS] is composed of the Low Head Safety Injection (LHSI) subsystem, the Medium Head Safety Injection (MHSI) subsystem and the accumulator subsystem, with the In-containment Refuelling Water Storage Tank (IRWST) shared by the three trains of the RIS [SIS].

The RIS [SIS] can be powered by the Emergency Diesel Generators (EDGs). Furthermore, the LHSI function can also be powered by the Station Black Out (SBO) diesel generators (for train A and train B).

A further description of the RIS [SIS] is given in Sub-chapter 7.5.

### 2.7.2 Emergency Boration System (RBS [EBS])

Under DBC-2/3/4 and DEC-A, the RBS [EBS] injects highly borated water into the RCP [RCS], via the RIS [SIS] cold leg injection line, to control the reactivity of the reactor during the transition from the controlled state to the safe state. The system would also be activated in case the control rods fail to insert under DEC-A.

The RBS [EBS] consists of three independent trains. Each train consists of one tank, one pump and associated pipes and valves. The pumps can be powered by the EDGs and SBO diesel generators.

A further description of the RBS [EBS] is given in Sub-chapter 7.6.

### 2.7.3 Atmospheric Steam Dump System (VDA [ASDS])

The VDA [ASDS] performs the function of removing the residual heat by discharging steam from SG into the atmosphere under DBC-2/3/4 and DEC-A.

The VDA [ASDS] consists of three independent trains aligned to the three SGs. Each train consists of one Main Steam Relief Isolation Valve (MSRIV), one Main Steam Relief Control Valve (MSRCV) and one silencer. The VDA [ASDS] is connected to the main steam line upstream of the Main Steam Isolation Valve (MSIV). When the valves are opened, the steam from the SGs is discharged to the atmosphere.

A further description of the VDA [ASDS] is given in Sub-chapter 7.7.

### 2.7.4 Emergency Feedwater System (ASG [EFWS])

The ASG [EFWS] provides emergency feedwater for the SGs to remove the residual heat when the normal feedwater systems are unavailable under DBC-2/3/4 and DEC-A.

The ASG [EFWS] consists of three identical trains corresponding to the three SGs. Each train consists of the following main equipment:

a) One storage tank;

b) One emergency feedwater pump;

c) Associated pipes and valves.

Each train of the ASG [EFWS] is located in one of the safeguard buildings. The pumps of each train can be powered by the EDGs. Furthermore, the pumps of train A and train B can also be powered by the SBO diesel generators.

A further description of the ASG [EFWS] is given in Sub-chapter 7.8.

### 2.7.5 Secondary Passive Heat Removal System (ASP [SPHRS])

The ASP [SPHRS] serves as the backup means of removing residual decay heat at the secondary side. When the ASG [EFWS] or VDA [ASDS] fails, it can be used for removing the residual heat of the reactor core from the secondary side passively during DEC-A. Moreover, the ASP [SPHRS] can provide makeup water to the spent fuel pool if all trains of the Fuel Pool Cooling and Treatment System (PTR [FPCTS]) cooling loop fail.

The ASP [SPHRS] is composed of three identical cooling trains, each serving one SG. Each train consists of one steam inlet pipe, one condenser, feed pipes and associated valves. The condenser is submerged in the water tank which is located on the outer wall of the reactor building.

The system operation is designed as passive operation using natural circulation. The cold heat sink, i.e. the water tank (where the condensers are submerged), is arranged at a higher position than the hot source (SG). The density difference can be formed between the steam pipe and the condensate pipe. Consequently, the height and density difference can provide a sufficient driving force for the natural circulation. The primary heat transferred to the SG secondary side can be removed by the water tank of the ASP [SPHRS] and finally discharged to the atmosphere by evaporating water in the tank.

A further description of the ASP [SPHRS] is given in Sub-chapter 7.9.

### 2.7.6 Containment Heat Removal System (EHR [CHRS])

The EHR [CHRS] functions under DEC, which acts as a diverse means for containment protection:

a) It limits the pressure of the containment with containment sprays, so as to maintain the integrity of the containment;

b) It injects enough water to the reactor pit to cool and assure the integrity of RPV during DEC-B;

c) It cools IRWST by transferring residual heat from IRWST to the ultimate heat sink.

The EHR [CHRS] consists of four subsystems: the containment spraying subsystem, the reactor pit injection subsystem, the strainer back-flushing subsystem and the passive reactor pit injection subsystem. Except for the passive reactor pit injection subsystem, each of the other subsystems has two identical trains which are physically separated. Each train consists of:

a) One intake line from the IRWST;

b) One containment heat removal pump;

c) One heat exchanger;

d) Three discharge lines (for containment spraying, reactor pit injection and strainer back-flushing);

e) Associated pipes and valves.

The passive reactor pit injection subsystem consists of one reactor pit flooding tank located inside the containment at an elevation higher than the reactor pit, and lines and valves at the outlet of the tank connected to the reactor pit.

The EHR [CHRS] is powered by the EDGs and SBO diesel generators and cooled by Extra Cooling System (ECS [ECS]) or Component Cooling Water System (RRI [CCWS]).

A further description of the EHR [CHRS] is given in Sub-chapter 7.4.2.

### 2.7.7 Extra Cooling System (ECS [ECS])

The ECS [ECS] removes the core residual heat and the decay heat from the spent fuel pool under DEC-A (such as Total Loss of Cooling Chain (TLOCC) and SBO) and DEC-B.

The ECS [ECS] consists of two identical trains. Each train consists of:

a) One ECS [ECS] terminal cooling loop, including a terminal circulation pump, a filter, a suite of mechanical draft cooling towers, relevant pipes, valves and pipe fittings;

b) One ECS [ECS] intermediate cooling loop, including an intermediate circulation pump, a heat exchanger, a surge tank, relevant pipes, valves and pipe fittings.

Water stored in the cooling tower pool is extracted by the terminal circulation pumps, conveyed to the cooling tower water distribution pipe through the heat exchanger, and sprayed via sprinklers.

The intermediate cooling loop is a closed loop. After discharge from the intermediate heat exchanger, the cooling water flows to the EHR [CHRS] heat exchanger in Safeguard Building A (BSA) or Safeguard Building B (BSB) and the Fuel Pool Cooling and Treatment System (PTR [FPCTS]) heat exchanger in the Fuel Building (BFX), and then flows back to the intermediate heat exchanger. The intermediate surge tank is connected to the inlet pipe of the intermediate circulation pump. Both trains of the ECS [ECS] are located in the Extra Cooling System and Fire-fighting Water Production System Building (BEJ), with separation within the building. The ECS [ECS] can be powered by the EDGs and the SBO diesel generators.

A further description of the ECS [ECS] is given in Sub-chapter 7.10.

### 2.7.8 Containment Filtration and Exhaust System (EUF [CFES])

The EUF [CFES] performs an active pressure relief function to decrease the pressure inside the containment and maintain the integrity of the containment during severe accidents.

A further description of the EUF [CFES] is given in Sub-chapter 7.4.3.

### 2.7.9 Containment Combustible Gas Control System (EUH [CCGCS])

The EUH [CCGCS] controls the hydrogen concentration in the containment to prevent a hydrogen explosion, thus ensuring the containment integrity during a loss of coolant accident or a severe accident.

The EUH [CCGCS] consists of the hydrogen recombination subsystem and the hydrogen monitoring subsystem:

a) The hydrogen recombination subsystem consists of 29 Passive Autocatalytic Recombiners (PARs) for DBC and DEC, which can reduce the hydrogen concentration passively once the concentration reaches the threshold value. The PARs which are located in the reactor building are entirely passive.

b) The hydrogen monitoring subsystem is composed of two redundant trains, which are located in the reactor building. Each train consists of five hydrogen sensors. The purpose is to provide information to operators about the hydrogen concentration in the containment building.

A further description of the EUH [CCGCS] is given in Sub-chapter 7.4.4.

## 2.8 Main Auxiliary Systems

The main auxiliary systems are the systems which are essential to maintain normal operation. The main auxiliary systems of the UK HPR1000 consist of nuclear auxiliary systems, process auxiliary systems, HVAC systems and fire protection systems. Brief summaries of the main auxiliary systems of the UK HPR1000 are provided below. A further description of these systems is given in Chapter 10.

### 2.8.1 Nuclear Auxiliary Systems

The nuclear auxiliary systems for the UK HPR1000 include Chemical and Volume Control System (RCV [CVCS]), Reactor Boron and Water Makeup System (REA [RBWMS]), Coolant Storage and Treatment System (TEP [CSTS]), Nuclear Sampling System (REN [NSS]), Fuel Pool Cooling and Treatment System (PTR [FPCTS]), Component Cooling Water System (RRI [CCWS]) and Essential Service Water System (SEC [ESWS]).

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

The RCV [CVCS] provides reactor coolant volume control, reactivity control, chemical control and reactor coolant pump seal injection functions during plant normal operation.

The RCV [CVCS] consists of the following subsystems:

a) Letdown subsystem;

b) Purification subsystem;

c) Volume Control Tank (VCT) and hydrogenation station;

d) Charging subsystem;

e) Reactor coolant pump seal water injection and leak-off recovery subsystem;

f) Chemical injection subsystem.

The RCV [CVCS] is mainly located in the Fuel Building (BFX) and the BRX, with the coolant purification unit being located in the Nuclear Auxiliary Building (BNX).

A further description of the RCV [CVCS] is given in Sub-chapter 10.4.3.

2.8.1.2 Reactor Boron and Water Makeup System (REA [RBWMS])

The REA [RBWMS] makes up boric acid solution or demineralised water to RCP [RCS] via the RCV [CVCS] to regulate the boron concentration of reactor coolant.

The REA [RBWMS] consists of boric acid mixing and distribution unit (REA 1), boric acid storage and injection units (REA 2 and REA 3) and demineralised water injection units (REA 4 and REA 5):

a) REA 1 prepares fresh boric acid for other units such as REA 2 and REA 3, PTR [FPCTS] and RBS [EBS].

b) REA 2 and REA 3 are redundant units, which provide boric acid to RCV [CVCS] to fulfil reactor coolant volume control and reactivity control functions.

c) REA 4 and REA 5 are also redundant units, which provide demineralised water to RCV [CVCS] to fulfil reactor coolant volume control and reactivity control functions.

The REA [RBWMS] is mainly located in the BNX.

A further description of the REA [RBWMS] is given in Sub-chapter 10.4.4.

2.8.1.3 Coolant Storage and Treatment System (TEP [CSTS])

The TEP [CSTS] receives, stores and treats reusable primary coolant during normal operation. It prepares demineralised water and 7000 mg/kg boric acid solution via the evaporation process and transfers them to storage tanks for reuse. Furthermore, the TEP [CSTS] reduces the radioactivity level in the primary coolant through the degasification process.

The TEP [CSTS] can be divided into the following subsystems:

a) Coolant storage and supply subsystem (TEP 1);

b) Coolant purification subsystem (TEP 2);

c) Coolant treatment subsystems (TEP 3, TEP 5 and TEP 6);

d) Coolant degasification subsystem (TEP 4).

The TEP [CSTS] is located in the BNX.

A further description of the TEP [CSTS] is given in Sub-chapter 10.4.5.

2.8.1.4 Nuclear Sampling System (REN [NSS])

The REN [NSS] is designed to monitor the physical and chemical characteristics of the primary systems via on-line devices and grab samples from the primary systems, secondary systems and other auxiliary systems.

A further description of the REN [NSS] is given in Sub-chapter 10.4.6.

2.8.1.5 Fuel Pool Cooling and Treatment System (PTR [FPCTS])

The PTR [FPCTS] consists of four subsystems: the Spent Fuel Pool (SFP) cooling unit, purification and water transfer unit, skimming unit and water make-up unit:

a) The SFP cooling unit has three redundant trains, and each train can cool the SFP. It operates as a closed-loop, with cooling being supplied via heat exchangers from the RRI [CCWS] and SEC [ESWS] under normal operation and DBC, or alternatively from ECS [ECS] (for train A and train B) under DEC-A. All three trains can be powered by the EDGs. Furthermore, train A and train B can also be powered by the SBO diesel generators. The three trains are all located in the fuel building, but are separated in separate rooms.

b) The purification and water transfer unit purifies the water in the BRX pools (including reactor cavity and internals storage compartment), BFX pools (including SFP, transfer compartment and cask loading pit), IRWST and reactor pit flooding tank.

c) The skimming unit cleans the water surface of the BRX pools and BFX pools.

d) The water make-up unit provides SFP make-up to compensate the loss of water during normal operation and accident conditions.

A further description of the PTR [FPCTS] is given in Sub-chapter 10.4.7.

2.8.1.6 Component Cooling Water System (RRI [CCWS])

The RRI [CCWS] removes heat (via heat exchangers) from safety-related and non-safety-related systems and equipment of the nuclear island. This system is a closed-loop cooling water system which delivers heat from its users to the SEC [ESWS] via heat exchangers.

The RRI [CCWS] consists of three safety-classified and separate trains. Each train is cooled by the Essential Service Water System (SEC [ESWS]) via the RRI [CCWS] heat exchanger. Train A and train B are of the same configuration with two pumps and two heat exchangers for each train. Train C is composed of one pump and one heat exchanger.

The RRI [CCWS] pumps can be powered by the EDGs. The main equipment of each train is arranged in the BSA, BSB and BSC respectively.

A further description of the RRI [CCWS] is given in Sub-chapter 10.4.8.

2.8.1.7 Essential Service Water System (SEC [ESWS])

The SEC [ESWS] fulfils the safety function of delivering the heat load collected by RRI [CCWS] to the heat sink (the sea) under DBC and DEC.

The SEC [ESWS] is composed of three separate and independent trains. Train A and train B are of the same configuration with two redundant sets of equipment for each train. Train C only has one set. Each set includes the following equipment:

a) Seawater filtering equipment;

b) Suction pipeline;

c) Essential service water pump;

d) Debris filter;

e) Discharge pipeline.

The SEC [ESWS] pumps are installed in two independent essential service water pumping stations. Train A and train C are located in the Essential Service Water Pumps Station-A (BPA) while being physically separated. Train B is located in the Essential Service Water Pumps Station-B (BPB). The SEC [ESWS] can be powered by the EDGs.

A further description of the SEC [ESWS] is given in Sub-chapter 10.4.9.

### 2.8.2 Process Auxiliary Systems

The process auxiliary systems of the UK HPR1000 include the following systems:

a) NI Dematerialised Water Distribution System (SED [DWDS (NI)]);

b) Potable Water System (SEP [PWS (NI)]);

c) Nuclear island gas distribution systems:

1) Oxygen Distribution System (SGO [ODS]);

2) Nitrogen Distribution System (SGN [NDS]);

3) NI Hydrogen Distribution System (SGH [HDS]);

d) Compressed air distribution systems.

A further description of these systems is given in Sub-chapter 10.5.

### 2.8.3 Heating, Ventilation and Air Conditioning (HVAC) Systems

The HVAC systems consist of three groups as follows:

a) Some of the ventilation systems play a direct role in supporting the radioactive confinement to limit the discharge of radioactive material into the environment under DBC and DEC. These systems include:

1) Nuclear Auxiliary Building Ventilation System (DWN [NABVS]);

2) Fuel Building Ventilation System (DWK [FBVS]);

3) Containment Internal Filtration System (EVF [CIFS]);

4) Containment Sweeping and Blowndown Ventilation System (EBA [CSBVS]);

5) Annulus Ventilation System (EDE [AVS]);

6) Safeguard Building Controlled Area Ventilation System (DWL [SBCAVS]);

7) Access Building Controlled Area Ventilation System (DWW [ABCAVS]);

8) Waste Treatment Building Ventilation System (DWQ [WTBVS]).

b) Some of the HVAC systems, as supporting systems, maintain ambient conditions (temperature, humidity and cleanliness) within the acceptable range for personnel and equipment, so as to ensure normal operation of the classified systems. These systems include:

1) Electrical Division of Safeguard Building Ventilation System (DVL [EDSBVS]);

2) Main Control Room Air Conditioning System (DCL [MCRACS]);

3) Diesel Building Ventilation System (DVD [DBVS]);

4) Fuel Building Ventilation System (DWK [FBVS]) (only heating in the classified boric acid room and cooling in some classified process system rooms);

5) Safeguard Building Controlled Area Ventilation System (DWL [SBCAVS]) (only heating in the classified boric acid room and cooling in some classified process system rooms);

6) Essential Service Water Pumping Station Ventilation System (DXS [ESWVS]);

7) Extra Cooling Water and NI Firefighting Building Ventilation System (DXE [ECW&FFB VS]);

8) Containment Cooling and Ventilation System (EVR [CCVS]).

c) The chilled water systems provide chilled water for the ventilation and air conditioning systems, these systems include:

1) Safety Chilled Water System (DEL [SCWS]), which supplies chilled water to the safety cooling coils of the DVL [EDSBVS], DWL [SBCAVS], DCL [MCRACS] and DWK [FBVS]. The DEL [SCWS] consists of three trains, of which one train is water-cooled by RRI [CCWS] and the other two trains are air cooled;

2) Operational Chilled Water System (DER [OCWS]), which supplies chilled water to the non-classified cooling coils of the DVL [EDSBVS], DWN [NABVS] and EVR [CCVS].

A further description of the HVAC systems is given in Sub-chapter 10.6.

### 2.8.4 Fire Protection Systems

The NI related fire protection systems are as follows:

a) Fire-fighting Water Production System (JAC [FWPS]);

b) Fire-fighting Water System for Nuclear Island (JPI [FWSNI]);

c) Fire Extinguishing System for Nuclear Island Diesel Generator Building (JPV [FSDB]);

d) Mobile and Portable Fire Extinguishing Equipment (JPS [MPFE]).

None of them contribute to the functions of reactivity control or residual heat removal. They also do not contribute to the function of confinement of radioactive substances, but containment isolation valves of JPI [FWSNI] contribute to ensure the containment integrity.

The JAC [FWPS] consists of two dedicated pools, three electrical fire pumps, pipes and valves. The JAC [FWPS] provides fire-fighting water to deal with the fire during the operation and shutdown of the plant, and the fire break out two weeks after earthquakes in the seismically qualified part of nuclear island.

The JPI [FWSNI] provides fire hose spray system and sprinkler systems for fires that may occur at buildings within the nuclear island.

The JPV [FSDB] provides fire hose and fixed water sprinkler-foam combination system in the diesel generator building, in which the open system is used to protect the main fuel tank room, while the closed system is used to protect the diesel generator room and the room for daily fuel tanks.

The JPS [MPFE] provides fire protection measures for the preliminary stage of a fire, for all buildings within the site by mobile and portable fire extinguishing equipment.

A further description of the fire protection systems is given in Sub-chapter 10.7.

## 2.9 Steam and Power Conversion Systems

The steam and power conversion systems are designed to remove heat from the RCP [RCS] via the three SGs and convert it to electric power through the turbine-generator. The main steam and power conversion systems of the UK HPR1000 are listed below. Brief introduction for important systems is given in this chapter. A further description of the steam and power conversion systems is given in Chapter 11.

### 2.9.1 Main Steam System (VVP [MSS])

The main function of the VVP [MSS] is to remove decay heat by transferring steam to the turbine generator set or the condenser. The VVP [MSS] consists of three main steam lines. Each line is connected to one SG and includes:

a) One Main Steam Isolation Valve (MSIV);

b) Two Main Steam Safety Valves (MSSVs);

c) One steam bypass line that bypasses the MSIV;

d) Drain lines;

e) Associated pipes and valves.

A further description of the VVP [MSS] is given in Sub-chapter 11.3.3.

### 2.9.2 Main Feedwater Flow Control System (ARE [MFFCS])

The ARE [MFFCS] regulates feedwater flowrate for supplying the SGs under normal conditions, and performs the feedwater line isolation function under accident conditions.

The ARE [MFFCS] consists of three main feedwater lines. Each line contains one Full Load Control Valve (FLCV), one Low Load Control Valve (LLCV), one Very Low Load Control Valve (VLLCV), one main isolation valve, two Full Load Isolation Valves (FLIVs) and two Low Load Isolation Valves (LLIVs).

The ARE [MFFCS] check valves and main feedwater lines which connect the SGs are independently arranged in the BRX.The other main components of the ARE [MFFCS] are located in the safeguard buildings in such a way that the three main feedwater lines are physically separated from each other.

A further description of the ARE [MFFCS] is given in Sub-chapter 11.3.4.

### 2.9.3 Steam Generator Blowdown System (APG [SGBS])

The APG [SGBS] treats a continuous blowdown flow in order to maintain the characteristics of the secondary-side water within predetermined chemistry limits during normal operation. During a Steam Generator Tube Rupture (SGTR) accident, the APG [SGBS] connecting lines between SGs are used to transfer radioactive water from the affected SG to the others.

The APG [SGBS] consists of a blowdown water collection unit, a blowdown cooling unit, a pressure reducing and flowrate controlling unit and a blowdown treatment unit.

A further description of the APG [SGBS] is given in Sub-chapter 11.3.5.

### 2.9.4 Non-Safety Related Steam and Power Conversion Systems

The non-safety related steam and power conversion systems include the Turbine Bypass System (GCT [TBS]), Feedwater Deaerating Tank and Gas Stripper System (ADG [FDTGSS]), Circulating Water System (CRF [CWS]), Condensate Polishing System (ATE [CPS]).

A further description of these systems is given in Sub-chapter 11.4.

## 2.10 Radioactive Waste Management Systems

The radioactive waste management systems include the liquid radioactive waste management systems, the gaseous radioactive waste management systems and the solid radioactive waste management system. Brief introduction of these systems is given below. A further description of the radioactive waste management systems is given in Chapter 23.

### 2.10.1 Liquid Radioactive Waste Management Systems

The liquid radioactive waste management systems are designed to control, collect, treat, transport and store the liquid radioactive wastes produced during the normal operation. The specific objectives of the liquid radioactive waste management systems are to:

a) Segregate liquid waste streams according to the waste categories, sources, and chemical properties;

b) Treat liquid waste;

c) Discharge the treated liquid wastes to the environment only when in compliance with the relevant discharge management objective and conditions.

The liquid radioactive waste management systems consist of the following subsystems:

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

b) Liquid Waste Treatment System (TEU [LWTS]);

c) Sewage Recovery System (SRE [SRS]);

d) Nuclear Island Liquid Waste Discharge System (TER [NLWDS]);

e) Conventional Island Liquid Waste Discharge System (SEL [LWDS (CI)]).

A further description of the liquid radioactive waste management systems is given in Sub-chapter 23.6.

### 2.10.2 Gaseous Radioactive Waste Management Systems

Gaseous radioactive waste which is generated unavoidably during the operation of UK HPR1000 is managed by gaseous radioactive waste management systems. It consists of the following systems:

a) Gaseous Waste Treatment System (TEG [GWTS]);

b) HVAC systems.

The TEG [GWTS] is designed to:

a) Flush the containers and tanks containing reactor coolant with nitrogen to avoid hydrogen accumulation in the gas space and limit the hydrogen/oxygen concentrations in the TEG [GWTS] and in flushed components to keep them under flammability limits;

b) Prevent radioactive gases escaping from the connected components into the building atmosphere by maintaining a slight negative pressure in the flushing section;

c) Collect and treat the excess gas flowrates arising from the connected components during plant start-up, shutdown or component flushing;

d) Delay the radioactive noble gases in the gaseous waste, to reach a discharge management objective, prior to discharging them into the environment.

A further description of the TEG [GWTS] is given in Sub-chapter 23.7.

The HVAC systems are briefly introduced in Sub-chapter 2.8.3.

### 2.10.3 Solid Radioactive Waste Management Systems

The function of the solid radioactive waste management systems is to manage the solid radioactive wastes and non-aqueous liquid wastes, including the collection, characterisation, segregation, treating, conditioning and storage of the wastes. The wastes include the spent resins, concentrates, spent filter cartridges, dry active wastes, sludge, oil and organic solvent, etc.

A further description of the solid radioactive waste management systems is given in Sub-chapter 23.8.

## 2.11 Instrumentation and Control

The I&C systems of the UK HPR1000 perform the functions of control, monitoring, protection and alarm to support the three fundamental safety functions.

A brief summary of the centralised I&C systems is provided below:

a) Plant Standard Automation System (PSAS) functions to monitor and control the plant in normal operation conditions. It performs the following main functions maintaining the main parameters of the plant within the normal operational range:

   1) Reactor coolant average temperature control;

   2) Reactor power control;

   3) Primary coolant inventory control and pressuriser level control;

   4) Pressuriser pressure control;

   5) Steam dump control;

   6) Steam generator level control, etc.

b) Protection System (PS) performs the functions of emergency reactor trip and safety systems actuation to bring the plant to the controlled state after DBC-2/3/4, including:

1) Reactor trip;

2) Turbine trip;

3) Safety injection;

4) ASG [EFWS] actuation and isolation;

5) VDA [ASDS] actuation and isolation, etc.

In addition, the PS transmits post-accidents monitoring signals to the auxiliary control panel, including steam generator level, pressuriser level and pressure, containment pressure, etc.

c) Safety Automation System (SAS) performs manual and automatic functions as well as the associated monitoring to bring the plant from the controlled state to the safe state after DBC-2/3/4:

1) The manual functions include manual starting and stopping of the safety injection pumps, manual stopping of the reactor coolant pumps, manual actuation of RBS [EBS], manual VDA [ASDS] opening and closing, etc.;

2) The automatic functions include automatic opening of the ASG [EFWS] flow control valve, automatic isolation of the reactor coolant pumps thermal barrier, etc.

SAS also performs the DEC-A features functions to mitigate the consequences of the failure of mechanical systems. The related systems include ASP [SPHRS], EHR [CHRS], ECS [ECS], etc.

d) Diverse Actuation System (KDS [DAS]) is designed to mitigate the consequence of DBC-2/3/4 with the concurrent common cause failure of PS and SAS, and to bring the plant to the final state.

e) Severe Accident I&C System (KDA [SA I&C]) provides monitoring and control functions to mitigate the consequence of DEC-B. KDA [SA I&C] performs the following functions:

1) Manual opening of the severe accident relief valves;

2) RCP [RCS] depressurisation;

3) Hydrogen monitoring;

4) Radioactivity monitoring;

5) Spent fuel pool monitoring, etc.

f) Control room system provides human machine interfaces with the operators to monitor and control the plant under all operating conditions and maintain the plant in a safe condition. The human machine interfaces are installed in Main

Control Room (MCR), remote shutdown station and technical support centre.

A brief summary of the main non-centralised I&C systems is provided below:

a) In-core Instrumentation System (RIC [IIS]) is designed to measure the temperature and reactor coolant level in RPV for normal and accident monitoring.

b) Nuclear Instrumentation System (RPN [NIS]) is designed to measure the reactor nuclear power uninterruptedly from start-up to full power operation of the reactor.

c) Plant Radiation Monitoring System (KRT [PRMS]) is designed to measure and indicate whether the radiological conditions in the plant are within the bounds of the designed conditions or not.

d) Rod Position Indication and Rod Control System (RGL [RPICS]) is designed to indicate the position of the control rod in the core and to control the reactor power and coolant temperature.

e) Nuclear Accident Emergency Management System (KCC [NAEMS]) is designed to provide the necessary information for emergency response of the plant.

f) Turbine Generator Control System (TGCS) is designed to perform the functions of turbine governing, protection and supervising.

A further description of the I&C systems is given in Chapter 8.

## 2.12 Electric Power

The electrical power system supports the Structures, Systems and Components (SSCs) to perform their required duties by providing reliable power supplies in multiple levels of defence in depth. It comprises both the off-site and on-site electrical power system. A brief introduction of them is given below, and further description of the electrical power system is given in Chapter 9.

### 2.12.1 Off-site Electrical Power System

The UK HPR1000 unit is connected to the external grid through a main connection and a standby connection.

For normal operations, the electric power generated by the plant is transmitted to the external grid through the main connection. The power can also be fed back from the external grid through the main connection.

The standby connection serves as a backup power supply of the main connection when the main connection fails.

### 2.12.2 On-site Electrical Power System

The on-site power system includes the following subsystems:

a) Normal Alternating Current (AC) power distribution system

The normal AC power distribution system consists of the following systems:

1) The NI 10 kV normal power distribution systems;

2) The Conventional Island (CI) 10 kV normal power distribution systems;

3) The NI 380 kV normal power distribution systems;

4) The CI 380 V normal power distribution systems.

The normal AC power distribution system is powered from the auxiliary transformer when the main connection or main generator is available or from the standby connection after power supply switchover. The normal AC power distribution system supplies power to equipment required for normal operation or shutdown of the unit, it also provides a power supply to the emergency AC power distribution system.

b) Emergency AC power distribution system

The emergency AC power distribution system consists of the following systems:

1) NI 10 kV emergency power distribution system;

2) NI 380 V emergency power distribution system.

The emergency AC power distribution system is normally supplied by the CI 10 kV normal power distribution system and can be supplied by the NI 10 kV emergency power supply system when the normal power distribution is unavailable. The emergency AC power distribution system supplies power to equipment required for continuous power supply under Loss of Offsite Power (LOOP) condition, it also provides power to the SBO AC power distribution system.

c) SBO AC power distribution system

The SBO AC power distribution system consists of the following systems:

1) NI 10 kV SBO power distribution system;

2) NI 380 V SBO power distribution system.

The SBO AC power distribution system is supplied by the 10 kV emergency power distribution system under normal or LOOP condition, and can be supplied by the NI 10 kV SBO power supply system when the emergency power distribution system is unavailable. The SBO AC power distribution system supplies power to equipment required for continuous power supply under SBO condition.

d) EDG

The EDGs provide backup power supplies to the essential loads (such as RIS

[SIS], RBS [EBS] and ASG [EFWS]) under LOOP conditions. Three redundant EDGs are designed according to the redundant trains of safety systems and are located in different emergency diesel generator buildings, physically separated from each other.

e) SBO diesel generator

The SBO diesel generators are designed to cope with SBO condition, which provide power supplies to the DEC loads. The two SBO diesel generators are physically separated and are located in different buildings.

f) Mobile diesel generator

The two mobile diesel generators are designed to cope with the condition that SBO coupled with SBO diesel generators failure, which provide temporary emergency power supply for accident mitigation.

g) CI Direct Current (DC) and AC Uninterruptible Power Supply (UPS) system

The CI DC and AC UPS system provides uninterruptible power supply to NI non-safety DC and uninterruptible loads, I&C equipment, electrical protection cubicles, switchgears, DC motors and also emergency lighting loads for plant operation and investment protection in the CI.

h) NI 2h DC and AC UPS system

The NI 2h DC and AC UPS system provides uninterruptible power supply to the safety classed I&C equipment, isolation valves, switchboards and lighting loads of MCR.

i) NI 12h DC and AC UPS system

The NI 12h DC and AC UPS system provides uninterruptible power supply to I&C equipment, isolation valves, switchboards and lighting loads of MCR. In addition, the NI 12h DC system also provides 220 V DC power to the control rod drive mechanisms.

## 2.13 Main Civil Structures

### 2.13.1 Layout Consideration

The UK HPR1000 adopts a single-unit layout in order to increase independence, and achieve better adaptability to local site conditions and the power grid. The general layout of the UK HPR1000 consists of the Nuclear Island (NI), the Conventional Island (CI) and the Balance of Plant (BOP).

In the design of the layout of UK HPR1000 civil structures, the following factors are mainly considered:

a) Radiation Protection

The NI buildings are divided into controlled areas and supervised areas by the strictly designed boundaries. Separated access routes are provided for the controlled areas and the supervised areas. The pathways in controlled area are shielded from the radioactive systems and components. The components that need daily maintenance, calibration, operation or check in controlled area are located at positions easy for access and with a relatively lower dose rate, while the components with high dose rate are arranged in separated compartments, and remote and mechanical measures are provided for operation, monitoring and maintenance.

b) Access

The layout design of the plant ensures the accessibility in regard to plant's construction, operation, maintenance and demolition activities. Inside the building, alternative pathways are designed to enhance the accessibility to the control functions that may require local manual intervention and evacuation during emergency situations.

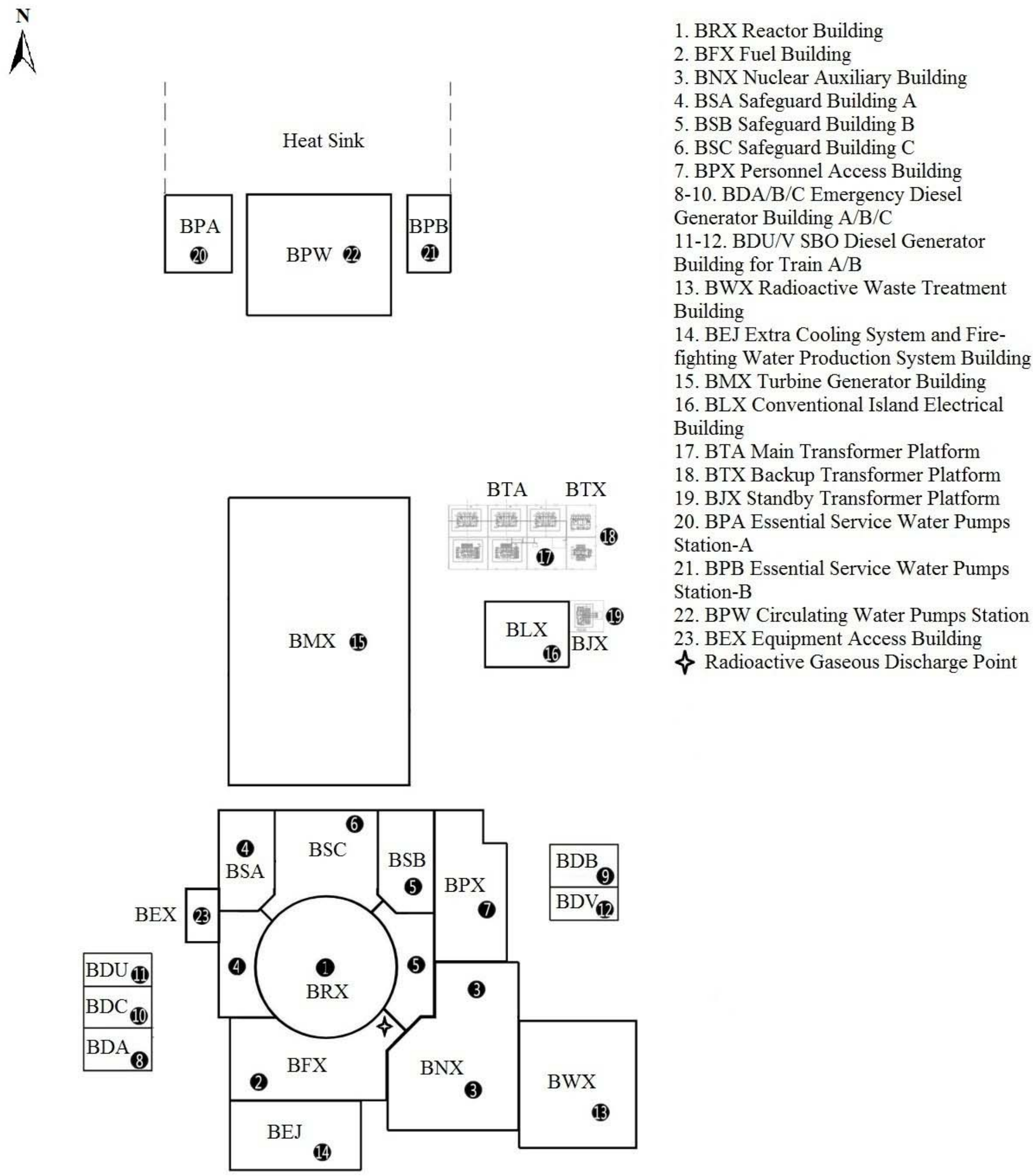
Pathways for equipment transportation inside the building are considered to facilitate the equipment's introduction and replacement. Adequate working space, hosting devices, removable shielding walls are provided to facilitate the maintenance and inspection of components. For components arranged above the floor, operational platforms are provided where necessary and practicable. Conventional safety requirements are also considered when locating the SSCs.

c) Minimise the adverse interaction in regard to internal and external hazards

The types of external and internal hazards and their assessments are described in Chapter 18 and Chapter 19. The plant layout considers the requirements for internal and external hazard protection. The buildings which contain redundant safety trains are divided into different divisions. Each division is separated with others by concrete walls and slabs, to prevent the internal hazards such as internal fire, flooding and explosion propagating from other trains. Other measures such as spatial separation and protective devices are also adopted to minimise the adverse interaction resulting from internal hazards.

The external hazard protection for SSCs inside the building is realised by the massive external walls and roofs. Spatial separation is also adopted for redundant buildings, to prevent losing more than one safety train during the same external hazard (e.g. aircraft crash).

F-2.13-1 shows the general layout of main buildings of the UK HPR1000. The detailed layout, especially the BOP buildings, depends on the site-specific characteristics and is not included in GDA.




F-2.13-1 General Layout of UK HPR1000

The NI includes the Reactor Building (BRX), three Safeguard Buildings (BSA/BSB/BSC), the Fuel Building (BFX), the Nuclear Auxiliary Building (BNX), the Emergency Diesel Generator Building A/B/C (BDA/BDB/BDC), the SBO Diesel Generator Building for Train A and for Train B (BDU/BDV), the Radioactive Waste Treatment Building (BWX).

The CI includes the Turbine Generator Building (BMX), the Conventional Island Electrical Building (BLX), the Main Transformer Platform (BTA), the Backup Transformer Platform (BTX) and the Standby Transformer Platform (BJX), but they are not included in the GDA scope according to Reference [2].

The most significant BOP structures include the Essential Service Water Pumps

Stations (BPA/BPB), the Extra Cooling System and Fire-fighting Water Production System Building (BEJ). Among them, only the BEJ is included in the GDA scope as described in Reference [2].

Brief introduction for the above buildings which are within the GDA scope is given below.

### 2.13.2 Reactor Building (BRX)

The Reactor Building (BRX) is mainly designed to arrange the primary coolant system. The containment of the reactor building acts as the third barrier during accident conditions, to prevent the unacceptable radioactive release to environment. The main SSCs arranged in BRX are presented in T-2.13-1.

BRX is located on the same raft foundation with Fuel Building (BFX) and Safeguard Buildings (BSA/BSB/BSC) and it is the centre of the nuclear island buildings. The vertical cross-section of BRX, BSC and BFX is shown in F-2.13-2.

BRX consists of internal containment, external containment and the internal structures (concrete and steel). The internal and external containments are cylindrical concrete structures and are separated by the annulus. The external containment is specially reinforced to withstand an aircraft crash, while the internal containment is pre-stressed and covered by a metallic leak tight liner on the inner surface.

The IRWST is located on the bottom of the building. The water tank of ASP [SPHRS], which is an annular concrete structure, is located around the top of the external containment. The reactor pit flooding tank is located next to the reactor pool and beneath the operational deck.

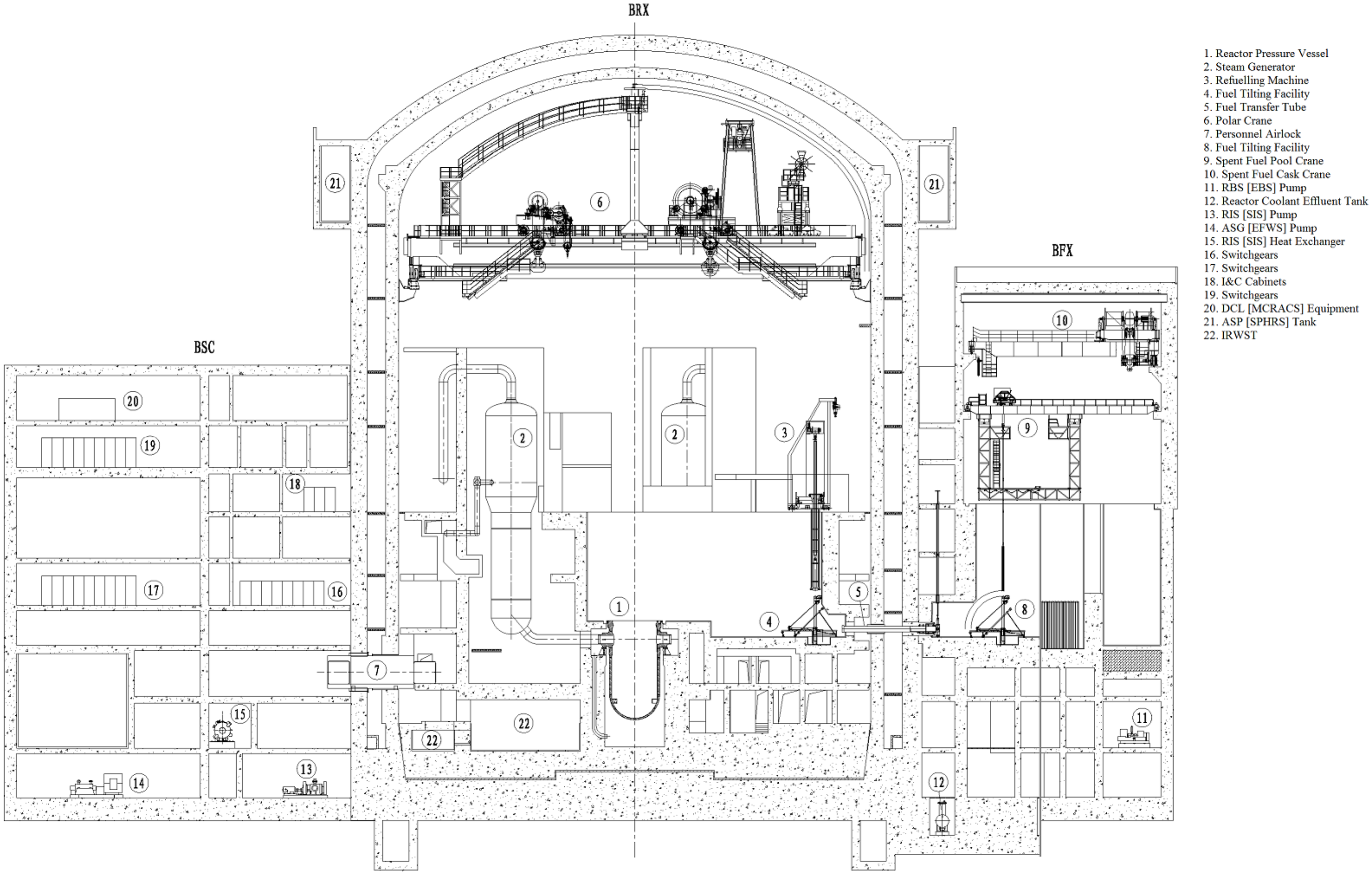
The three primary loops are arranged within the internal containment and enclosed by the secondary shielding wall. Each loop is separated from the others by massive walls, in order to avoid common cause failure, as well as to reduce the radiation dose to the outside area.

The space between secondary shielding wall and internal containment is the annular space, which serves as equipment transportation and a personnel circulation pathway. Main and auxiliary hoisting holes are designed in intermediate floors of the annular space to facilitate equipment introduction and replacement.

The equipment hatch is located on the operational deck (+17.50 m). The personnel airlock is located on +1.20 m floor and connected to BSC. The emergency personnel airlock is located on the operational deck and connected to the BFX. A main, and an auxiliary staircase, are included within the BRX. The main staircase is close to the personnel airlock while the auxiliary staircase is close to the emergency personnel airlock. For both staircases, an exit to the annular space on each floor is designed. This design feature provides alternative access and an evacuation pathway inside the BRX.

T-2.13-1 Main SSCs Arranged in BRX

| No. | SSCs |
|---|---|
| 1 | Main Feedwater Flow Control System (ARE [MFFCS]) |
| 2 | Secondary Passive Heat Removal System (ASP [SPHRS]) |
| 3 | Containment Heat Removal System (EHR [CHRS]) |
| 4 | Containment Combustible Gas Control System (EUH [CCGCS]) |
| 5 | Reactor Coolant System (RCP [RCS]) |
| 6 | Chemical and Volume Control System (RCV [CVCS]) |
| 7 | Safety Injection System (RIS [SIS]) |
| 8 | Main Steam System (VVP [MSS]) |
| 9 | Fuel Handling and Storage System (PMC [FHSS]) |
| 10 | In-containment Refuelling Water Storage Tank (IRWST) |
| 11 | Reactor pit |
| 12 | Reactor pit flooding tank |
| 13 | Polar crane |
| 14 | Equipment hatch |
| 15 | Personal airlocks |




F-2.13-2 Vertical Cross-section of BRX, BSC and BFX

### 2.13.3 Fuel Building (BFX)

BFX is located to the south of BRX and shares the same raft foundation with the BRX and the safeguard buildings. BFX is used to house the SSCs for the reception, storage, lifting, transfer and cooling of the fuel. In addition, BFX is also used to arrange some other nuclear island fluid systems and their support systems. The main SSCs arranged in BFX are presented in T-2.13-2.

In the BFX, the area from the bottom to the ground floor houses the process systems. This area is divided into three independent zones, to ensure the physical separation of the systems arranged in this area. High radioactive systems are arranged in the area closed to BRX and shielded from the pathway.

The area above the ground floor is mainly used to arrange the Fuel Handling and Storage System (PMC [FHSS]). The spent fuel pool, fuel reception compartment, new fuel assembly storage room, fuel operational hall and cranes are arranged in this area.

The external walls and roof of BFX are reinforced to protect the spent fuel pool against external hazards such as an aircraft crash.

The operators can enter the BFX through the entrance connected to BSA, BSB and BNX on the ground floor, and then access each floor via the lifts or staircases which are located on both sides of BFX. Two emergency exits are designed, one is on the ground floor leading directly outside, and the other is on -4.90 m floor leading to the BSB. Alternative entrance routes are designed into the BFX according to the size of equipment.

T-2.13-2 Main SSCs Arranged in BFX

| No. | SSCs |
|---|---|
| 1 | Spent fuel pool |
| 2 | Cask wash-down pit |
| 3 | Cask loading pit |
| 4 | Fuel transfer pit |
| 5 | Fuel Handling and Storage System (PMC [FHSS]) |
| 6 | Fuel Pool Cooling and Treatment System (PTR [FPCTS]) |
| 7 | Emergency Boration System (RBS [EBS]) |
| 8 | Chemical and Volume Control System (RCV [CVCS]) |

### 2.13.4 Safeguard Buildings (BSA/BSB/BSC)

The safeguard buildings are used to house the systems performing safety functions, the main control room and other supporting systems. The main SSCs arranged in the safeguard buildings are presented in T-2.13-3.

The safeguard buildings share the same raft foundation with BRX and BFX and include three completely independent sub-buildings (namely BSA, BSB and BSC), corresponding to the three trains of safety systems respectively. BSA and BSB are located on the opposite sides of the BRX and are spatially separated. BSC is between BSA and BSB. BSA/BSB/BSC are physically separated with each other by massive walls, to ensure that no more than one train of the safety systems inside the safeguard buildings is lost after a postulated internal or external hazard.

The main control room is located in the BSC, which is protected against aircraft crash by the reinforced designed external wall (and roof) of the BSC and the adjacent buildings of BSA, BSB and BRX.

The safeguard buildings can be divided into mechanical area and electrical, I&C and HVAC area vertically. The area from bottom to the ceiling of +4.90 m floor is the mechanical area, in which the safety fluid process systems (e.g. ASG [EFWS] and RIS [SIS]) are located. The area from +4.90 m to the roof of the building is electrical, I&C and HVAC area, in which the safety related I&C and HVAC systems, main control room and other supporting systems are located.

The safeguard buildings are divided into controlled area and supervised area with strictly designed boundaries. The controlled area where the radioactive systems are installed is located in the mechanical area and close to the BRX.

Individual pathways and staircases are designed in both the controlled and supervised areas. The controlled areas of BSA/BSB/BSC are interconnected on ground floor (+0.00 m), including connection to the controlled area of BRX, BFX and BNX, while the supervised areas of BSA/BSB/BSC are interconnected on +8.70 m floor.

In each area (building), the controlled and supervised areas are equipped with hoisting holes to facilitate the equipment introduction and replacement.

T-2.13-3 Main SSCs Arranged in the Safeguard Buildings

| No. | SSCs |
|---|---|
| 1 | Main Control Room (MCR) |
| 2 | Main Feedwater Flow Control System (ARE [MFFCS]) |
| 3 | Emergency Feedwater System (ASG [EFWS]) |
| 4 | Containment Heat Removal System (EHR [CHRS]) |
| 5 | Safety Injection System (RIS [SIS]) |
| 6 | Component Cooling Water System (RRI [CCWS]) |

| No. | SSCs |
|---|---|
| 7 | Atmospheric Steam Dump System (VDA [ASDS]) |
| 8 | Main Steam System (VVP [MSS]) |
| 9 | Main Control Room Air Conditioning System (DCL [MCRACS]) |

### 2.13.5 Nuclear Auxiliary Building (BNX)

The BNX is designed to house the systems which are necessary for plant normal operation. The main systems arranged in BNX are presented in T-2.13-4.

BNX is located on a separate raft foundation and is bordered on BFX and BSB.

The area close to the BFX is arranged with TEP [CSTS] and TEG [GWTS]. The centralised sampling equipment is arranged at the lowest level of BNX and close to BSB. The filter units of DWN [NABVS] are arranged at the upper levels of BNX.

A “hot workshop” room is designed within the controlled area of BNX, which is used for the maintenance of some components in the controlled area of NI buildings.

The north part of the BNX is the supervised area, where DER [OCWS], electrical power distribution systems and related supporting systems are arranged.

Individual personal circulation aisles are designed in the controlled and supervised areas respectively, and emergency exits are also provided for evacuation.

Individual equipment entrance routes are designed for the controlled and supervised areas respectively. The equipment can be brought to the dedicated rooms via the hoisting holes and lifts designed in this building.

T-2.13-4 Main Systems arranged in BNX

| No. | Systems |
|---|---|
| 1 | Coolant Storage and Treatment System (TEP [CSTS]) |
| 2 | Gaseous Waste Treatment System (TEG [GWTS]) |
| 3 | Reactor Boron and Water Makeup System (REA [RBWMS]) |
| 4 | Nuclear Sampling System (REN [NSS]) |
| 5 | Steam Generator Blowdown System (APG [SGBS]) |
| 6 | Nuclear Auxiliary Building Ventilation System (DWN [NABVS]) |
| 7 | Operational Chilled Water System (DER [OCWS]) |

### 2.13.6 Radioactive Waste Treatment Building (BWX)

The BWX is designed to arrange the reception and treatment facilities for radioactive waste from the operational and maintenance activities in the nuclear island.

The main systems installed in BWX are:

a) Solid Waste Treatment System (TES [SWTS]);

b) Liquid Waste Treatment System (TEU [LWTS]);

c) Hot Laundry System (SBE [HLS]);

d) Sewage Recovery System (SRE [SRS]);

e) Radioactive Decontamination System (SBD [RDS]);

f) Waste Treatment Building Ventilation System (DWQ [WTBVS]).

### 2.13.7 Emergency Diesel Generator Buildings and Station Black Out Diesel Generator Buildings

For UK HPR1000, each unit is designed with three emergency diesel generator buildings (BDA/BDB/BDC) and two SBO diesel generator buildings (BDU/BDV), which form two groups of buildings. The two groups of buildings are arranged on both sides of the reactor building separately. One group of buildings include two emergency diesel generator buildings (BDA and BDC) and one SBO diesel generator building (BDU). The other group consists of one emergency diesel generator building (BDB) and one SBO diesel generator building (BDV). The diesel generator buildings in each group are physically separated and constructed with reinforced concrete. One diesel generator and its auxiliary systems and equipment are located in each diesel generator building.

### 2.13.8 Extra Cooling System and Fire-fighting Water Production System Building (BEJ)

The BEJ is located adjacent to the south side of BFX, which is designed to house ECS [ECS] and JAC [FWPS].

The pools of ECS [ECS] and JAC [FWPS] are located in the centre, and two mechanical cooling towers are arranged on both sides of this building. The area between the pools and the cooling towers of BEJ are arranged with the equipment of ECS [ECS] and JAC [FWPS].

## 2.14 Spent Fuel Interim Storage (SFIS)

In the UK HPR1000 design, after storage for several years in the Spent Fuel Pool (SFP), the spent fuel is loaded into a transfer cask/canister and moved to a spent fuel interim storage facility for storage, prior to retrieval and repackaging for final disposal off-site.

A further description of the spent fuel interim storage is given in Chapter 29.

## 2.15 Emergency Preparedness

The on-site facilities used for emergency preparedness are equipped in UK HPR1000, including main control room, remote shutdown station, on-site emergency control centre, technical support centre, operation support centre, public information centre and communication systems.

The further description of emergency preparedness is given in Chapter 32.

## 2.16 Concluding Remarks

This chapter presents an overview of the UK HPR1000. It includes the UK HPR1000 evolution process, the main technical characteristics, the summary of system configuration, the main civil structures and other plant information.

The UK HPR1000 takes HPR1000 (FCG3) as the reference design. The development process of the HPR1000 (FCG3), which started from the import of M310 and went through CPR1000, CPR1000$^+$ and ACPR1000, was based on relevant good international practice demonstrating a continuous improvement of safety performance.

Based on the reference design, the design of UK HPR1000 is developed considering the requirements of UK context and RGP. In the design process of the UK HPR10000, the improvements will be continuously identified and incorporated.

## 2.17 References

[1] CGN, HPR1000 R&D History, GHX99980001DXZJ01MD, Rev. A, August 2018.

[2] GNS, Scope for UK HPR1000 GDA Project, HPR/GDA/REPO/0007, Rev. 000, May 2018.