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Pre-Construction Safety Report

Chapter 16

Civil Works & Structures

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16.1 List of Abbreviations and Acronyms

ALARP	As Low As Reasonably Practicable
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
BGA	Essential Service Water Supply Gallery A
BGB	Essential Service Water Supply Gallery B
BGC	Essential Service Water Supply Gallery C
BGH	Diesel Buildings Integrated Gallery H
BGI	Diesel Buildings Integrated Gallery I
BGJ	Diesel Buildings Integrated Gallery J
BGL	Essential Service Water Drain Gallery L
BGM	Essential Service Water Drain Gallery M
BGN	Essential Service Water Drain Gallery N
BMX	Turbine Generator Building
BNX	Nuclear Auxiliary Building
BPA	Essential Service Water Pumping Station A
BPB	Essential Service Water Pumping Station B
BPW	Circulating Water Pumping Station
BPX	Personnel Access Building
BRX	Reactor Building
BSA	Safeguard Building A

BSB	Safeguard Building B
BSC	Safeguard Building C
BWX	Radioactive Waste Treatment Building
CDM	Construction (Design and Management)
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DR	Design Reference
EAU	Containment Instrumentation System
EMIT	Examination, Maintenance, Inspection and Testing
EUR	European Utility Requirement
FE	Finite Element
FRS	Floor Response Spectra
GDA	Generic Design Assessment
HPR1000 (FCG3)	Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3
HVAC	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
I&C	Instrumentation & Control
LLSF	Low Level Safety Function
LOCA	Loss of Coolant Accident
NDT	Non-Destructive Testing
NI	Nuclear Island
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (US)
ONR	Office for Nuclear Regulation (UK)
OPEX	Operating Experience
PCSR	Pre-Construction Safety Report
PSA	Probabilistic Safety Assessment

RCP [RCS]	Reactor Coolant System
RGP	Relevant Good Practice
RO	Regulatory Observation
RP	Requesting Party
RQ	Regulatory Query
SAPs	Safety Assessment Principles
SFIS	Spent Fuel Interim Storage
SFP	Spent Fuel Pool
SFR	Safety Function Requirement
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
TAGs	Technical Assessment Guides
UK HPR1000	UK version of the Hua-long Pressurised Reactor
V&V	Verification and Validation

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

16.2 Introduction

This Chapter is part of the Pre-Construction Safety Report (PCSR) which is submitted to the Office for Nuclear Regulation (ONR) under the Generic Design Assessment (GDA) process. The PCSR demonstrates that the design, construction, operation and decommissioning of the UK version of the Hua-long Pressurised Reactor (UK HPR1000) are safe, secure and protect people and the environment.

The purpose of this Chapter is to present the contents of the civil structures design of the UK HPR1000. The safety case, including the relevant supporting documents, is described in this Chapter to underpin the fundamental objective of the UK HPR1000 which is described in Chapter 1 Sub-chapter 1.6.2.

The current civil engineering safety case is produced based on the version 3 of the UK HPR1000 Design Reference (DR3), as described in *UK HPR1000 Design Reference Report*, Reference [1]. The Civil Engineering area adopts the measures listed below to ensure consistency between the civil engineering safety documents and DR latest state.

a) The finite element modelling and calculations for buildings are originally undertaken based on DR version 1.0.

b) Each design change triggered after DR version 1.0 up to the latest DR version has been tracked and evaluated. During the circulation of the design changes, the design changes which have an impact on the civil structures are identified and incorporated into the civil engineering safety case documents.

16.2.1 Scope

This Sub-chapter provides the scope of UK HPR1000 PCSR Chapter 16 assessed during GDA. The scope includes the building scope and extent of scope which are consistent with the *Scope for UK HPR1000 GDA Project*, Reference [2].

16.2.1.1 Building Scope

The structures in scope for assessment during GDA are comprised of:

a) Safety Class 1 and 2 structures and/or Seismic Category 1 (SSE1) and 2 (SSE2) structures that are independent of the site specific conditions.

b) Site specific buildings that are important to nuclear safety.

c) Non-classified buildings where collapse could affect Class 1 and Class 2 structures.

The generic site layout for the UK HPR1000 is based on a single unit reactor design, which is shown in F-16.2-1. The overall view is shown in F-16.2-2.

The structures which are within the scope of GDA include the:

a) Reactor Building (BRX), which includes BRX internal structures, internal containment and external containment.

b) Safeguard buildings, which include Safeguard Building A (BSA), Safeguard Building B (BSB) and Safeguard Building C (BSC).

c) Fuel Building (BFX).

d) Nuclear Auxiliary Building (BNX).

e) Diesel generator buildings, which include Emergency Diesel Generator Building A (BDA), Emergency Diesel Generator Building B (BDB), Emergency Diesel Generator Building C (BDC), SBO Diesel Generator Building for Train A (BDU) and SBO Diesel Generator Building for Train B (BDV).

f) Radioactive Waste Treatment Building (BWX).

g) Extra Cooling System and Fire-fighting Water Production System Building (BEJ).

h) Equipment Access Building (BEX).

i) Personnel Access Building (BPX).

The structures which are outside the scope of GDA include the:

a) Essential Service Water Pumping Station A (BPA) and Essential Service Water Pumping Station B (BPB).

b) Circulating Water Pumping Station (BPW).

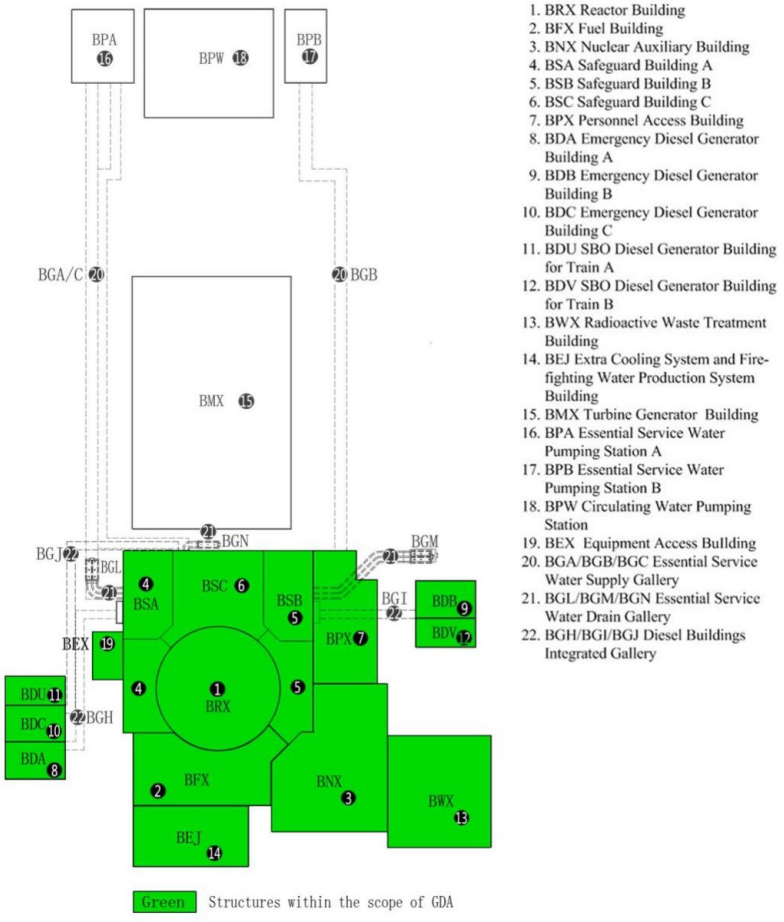
c) Essential Service Water Supply Gallery A (BGA), Essential Service Water Supply Gallery B (BGB) and Essential Service Water Supply Gallery C (BGC).

d) Diesel Buildings Integrated Gallery H (BGH), Diesel Buildings Integrated Gallery I (BGI) and Diesel Buildings Integrated Gallery J (BGJ).

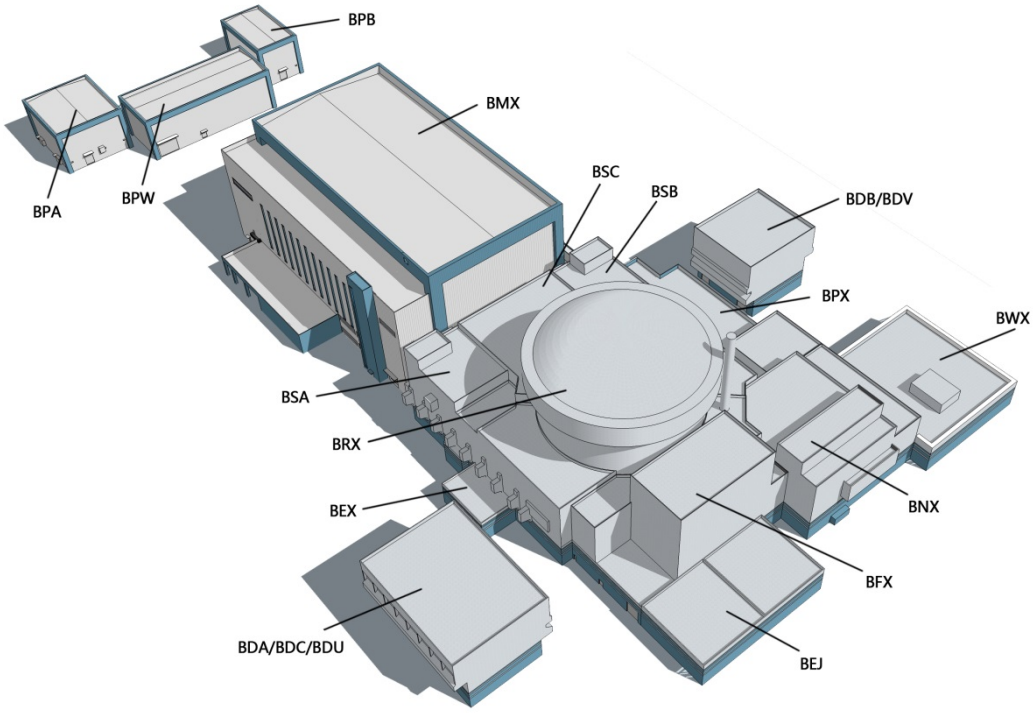
e) Essential Service Water Drain Gallery L (BGL), Essential Service Water Drain Gallery M (BGM) and Essential Service Water Drain Gallery N (BGN).

f) Turbine Generator Building (BMX).

g) Other buildings.



F-16.2-1 Generic Site Layout of the UK HPR1000



F-16.2-2 Overall View of the UK HPR1000

16.2.1.2 Extent of Scope for GDA

The extent of scope for GDA is proportional to the classification of the buildings. For safety class structures, a complete safety case is provided based on the generic site parameters which includes safety functional requirements and engineering requirements, seismic analysis, structural analysis and design, beyond design basis evaluation, design substantiation, construction, Examination, Maintenance, Inspection and Testing (EMIT), As Low As Reasonably Practicable (ALARP) assessment, etc. For non-classified structures, assessment that demonstrates these structures do not impact or adversely affect the safety classified structures is provided.

Detailed design documents that have to take into account site characteristics, such as the settlement, soil bearing capacity, and soil nonlinear behaviour will not be provided for GDA. Detailed construction specifications are site specific and will not be addressed in GDA.

The buildings outside the GDA scope include BMX, galleries (BGA/BGB/BGC, BGL/BGM/BGN, and BGH/BGI/BGJ), pumping stations (BPA, BPB, BPW) and other buildings. In the site specific stage, relevant measures will be taken to ensure that the buildings outside the GDA scope will not adversely interact with buildings within the GDA scope.

16.2.2 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 and a number of level 2 claims are developed and presented in Chapter 1 Sub-chapter 1.6. This chapter supports the ***Claim 3.3*** derived from the high level ***Claim 3***.

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

In terms of civil structures, ***Claim 3.3.9*** is developed according to the ***Claim 3.3.***

Claim 3.3.9*: The design of the civil works and structures has been substantiated.*

To support ***Claim 3.3.9***, this Chapter develops five Sub-claims and a number of relevant arguments and pieces of evidence shown in Appendix 1 table T-16.13-1.

16.2.3 Chapter Structure

The general structure of this Chapter is presented as the following:

a) Sub-chapter 16.1 List of Abbreviations and Acronyms

   This sub-chapter lists the abbreviations and acronyms that are used in this Chapter.

b) Sub-chapter 16.2 Introduction

   This sub-chapter presents the sub-claims, arguments and evidence of the civil structures design, and describes the scope of the PCSR Chapter 16 and interfaces with other Chapters.

c) Sub-chapter 16.3 Safety Functional Requirements and Engineering Requirements

   This sub-chapter describes the safety functional requirements, classification and categorisation and engineering requirements for the civil structures.

d) Sub-chapter 16.4 Applicable Codes and Standards

   This sub-chapter presents the selection approach and list of the applicable codes and standards.

e) Sub-chapter 16.5 Seismic Analysis

   This sub-chapter presents the seismic input for SSE1 and SSE2 structures,

modelling of structures, seismic analysis of SSE1 and SSE2 structures, and analysis of non-nuclear seismic structures.

f) Sub-chapter 16.6 Design of Nuclear Safety-related Structures

This sub-chapter describes the design of nuclear safety-related structures which includes structure description, design principles, loads and load combinations, methods of analysis and design, and substantiation.

g) Sub-chapter 16.7 Beyond Design Basis Conditions

This sub-chapter presents the beyond design basis conditions, which include ultimate capacity evaluation for internal containment, aircraft impact evaluation and cliff-edge effects of extreme environmental hazards for civil engineering structure.

h) Sub-chapter 16.8 Construction and Testing

This sub-chapter summarises the constructability of buildings and testing to be undertaken during construction.

i) Sub-chapter 16.9 In-service Inspection and Testing

This sub-chapter describes the building instrumentation, structural integrity test of the containment, containment leak-tightness testing, and routine inspection and maintenance.

j) Sub-chapter 16.10 ALARP

This sub-chapter presents the holistic and specific ALARP assessment and provides the ALARP conclusion.

k) Sub-chapter 16.11 Concluding Remarks

This sub-chapter presents concluding remarks for this chapter.

l) Sub-chapter 16.12 References

This sub-chapter lists the supporting references for this chapter.

m) Sub-chapter 16.13 Appendix 1 Route Map of PCSR Chapter 16

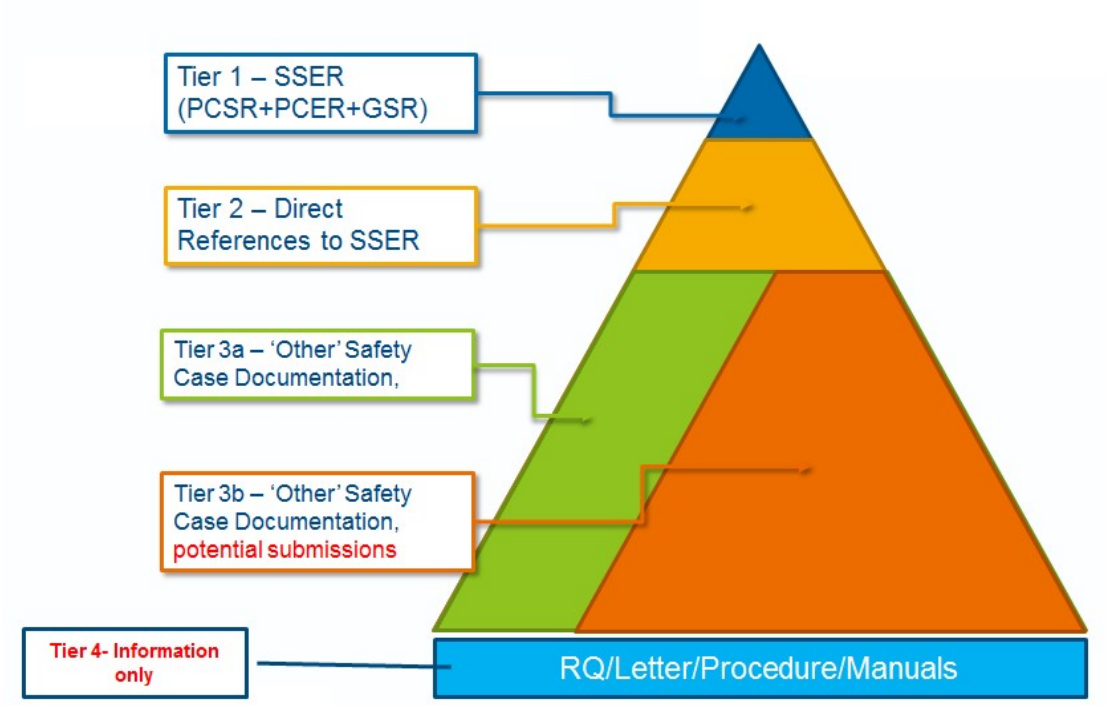
This sub-chapter provides the detailed Claims (Sub-claims), Arguments and Evidences for PCSR Chapter 16.

n) Sub-chapter 16.14 Appendix 2 Screen of the LLSFs relevant to Civil Engineering

This sub-chapter provides the detailed screening process of the Low Level Safety Functions (LLSFs) relevant to civil engineering.

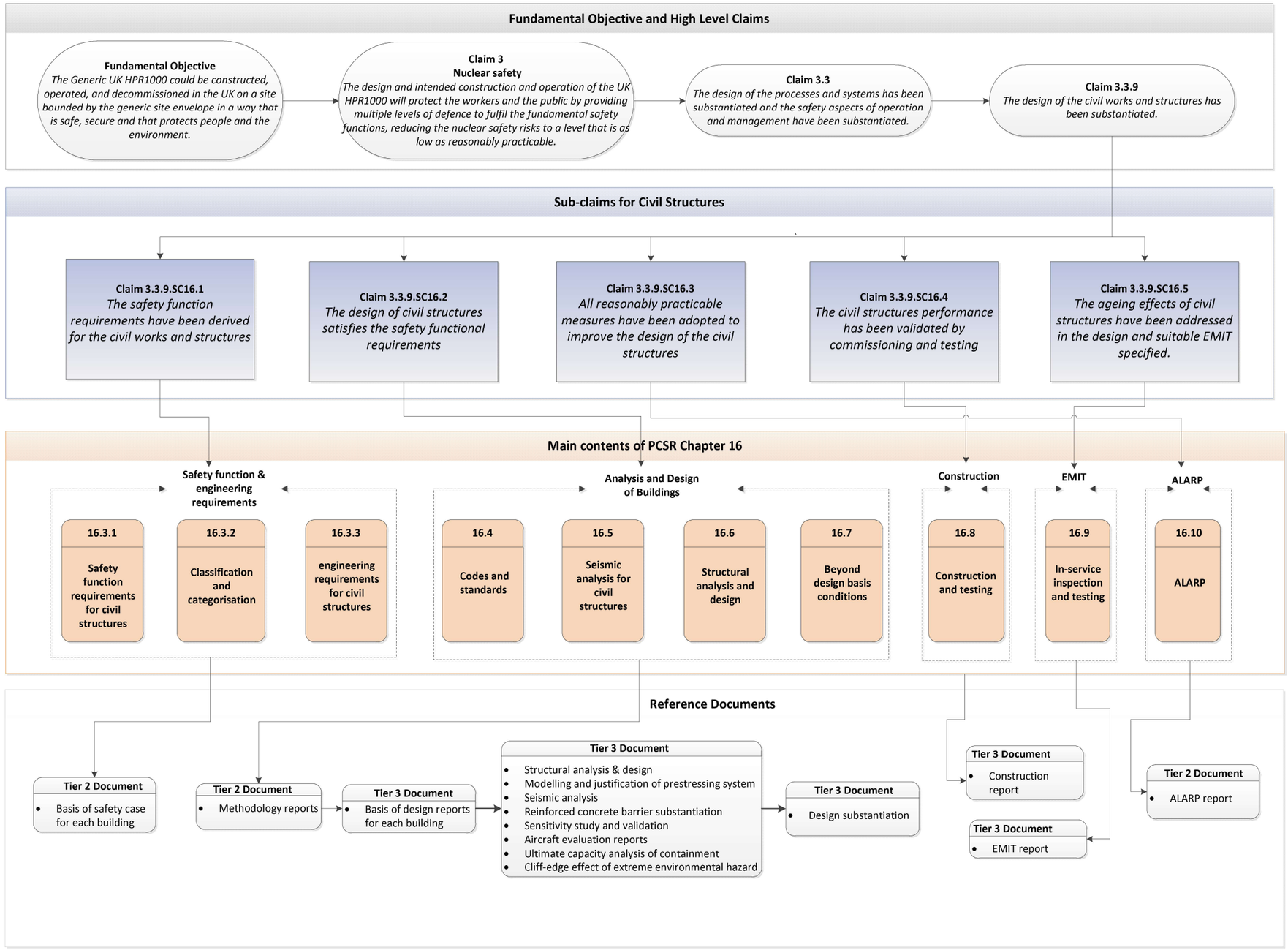
This chapter and relevant reference documents form the complete safety case for the civil structures. The PCSR presents the Claims and Sub-claims for the civil structures

(Tier 1 document). The basis of safety case reports, methodology reports, the generic design parameters report and other reports which are direct references to the PCSR present the arguments (Tier 2 documents); other safety case documents (Tier 3a documents are submitted to ONR, and the Tier 3b documents act as the potential submissions) provide the evidence; and the responses to the Regulatory Query (RQ) / Regulatory Observation (RO), correspondence, presentations, meeting minutes, etc., are for information only (Tier 4 documents). The general documentation hierarchy of civil engineering is shown in Figure F-16.2-3.



F-16.2-3 General Documentation Hierarchy of Civil Engineering

The main contents of this chapter are divided into five aspects which are safety functional requirements and engineering requirements, analysis and design of the buildings, ALARP, construction and EMIT. The clear traceability is developed based on the main contents as shown in F-16.2-4.



F-16.2-4 Structure of Chapter 16

16.2.4 Interfaces with Other Chapters

The interfaces with other chapters are listed in Table T-16.2-1.

T-16.2-1 Interfaces between Chapter 16 and Other Chapters

Chapter	Interface
Chapter 1 Introduction	Chapter 1 provides the fundamental objective, level 1 claims and level 2 claims. Chapter 16 provides chapter claims, arguments to support relevant claims that are addressed in Chapter 1.
Chapter 2 General Plant Description	Chapter 2 provides an introduction to the main civil structures and their layouts. Chapter 16 provides a further description of the civil works and structures mentioned in Sub-chapter 2.13.
Chapter 3 Generic Site Characteristics	Chapter 3 provides the values for site related parameters for the UK HPR1000 design applied in Chapter 16 for Civil Works and Structures design and presents a preliminary description of seismic and soil characteristics. Chapter 16 presents the design of civil works and structures which is based on these envelope values.
Chapter 4 General Safety and Design Principles	Chapter 4 provides general engineering substantiation principles, general principles for application of codes and standards, etc. Chapter 16 presents the design of civil works and structures which is based on these principles.
Chapter 6 Reactor Coolant System	The Chapter 6 provides the design of the RCP [RCS] system and component. Chapter 16 provides design information relevant to the reactor building that houses the RCP [RCS] system and component.
Chapter 7 Safety Systems	Chapter 7 provides the description of the containment general functional design. Chapter 16 provides the detailed description of the

Chapter	Interface
	containment structure.
Chapter 10 Nuclear Auxiliary Systems	Chapter 10 provides detailed design information for the auxiliary systems and equipment. Chapter 16 provides the output of civil structures (e.g. floor response spectrum, different displacement, etc.) that shall be considered in the auxiliary system and component design.
Chapter 11 Steam and Power Conversion System	Chapter 11 provides detailed design information for the steam and power conversion system. Chapter 16 provides the design output of civil structures (e.g. floor response spectrum, different displacement, etc.) that shall be considered in the steam and power conversion system design.
Chapter 14 Probabilistic Safety Assessment	Chapter 14 provides the risk insights to support potential improvements of the containment design. Chapter 16 provides design information for structures to support Probabilistic Safety Assessment (PSA).
Chapter 17 Structural Integrity	Chapter 17 presents the structural integrity classification and demonstration of the containment steel liner. Chapter 16 provides the relevant information for the containment steel liner.
Chapter 18 External Hazards	Chapter 18 defines the external hazards that lead to loads being placed on the civil structures. Chapter 16 presents the design substantiation of the civil structures.
Chapter 19 Internal Hazards	Chapter 19 defines the internal hazards (e.g. fire, explosion, flooding) to be considered in the substantiation of the civil structures. Chapter 16 presents the design substantiation of the civil structures.

Chapter	Interface
Chapter 20 MSQA and Safety Case Management	Chapter 20 presents Safety Case and Design Control Management including the relevant requirements, processes and coding system for Requirements Management. Chapter 16 applies the arrangements of Requirements Management set out in Chapter 20 [note] .
Chapter 23 Radioactive Waste Management	Chapter 23 presents the design of the radioactive waste management systems. Chapter 16 provides design information relevant to the buildings that house the radioactive waste management systems.
Chapter 24 Decommissioning	Chapter 24 presents the principles of building and structure design that facilitate the decommissioning and demolition of structures. Chapter 16 covers the requirements of civil works and structures to be applied for 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 civil works and structures. Chapter 16 provides the design information to demonstrate the conventional health and safety risk management techniques and general prevention principles are applied in the design process of the civil works and structures.
Chapter 28 Fuel Route and Storage	Chapter 28 presents the design of the PMC [FHSS] system and components. Chapter 16 provides the design substantiation for the BRX and BFX buildings that house the PMC [FHSS].
Chapter 29 Interim Storage of Spent Fuel	Chapter 29 presents the Spent Fuel Interim Storage (SFIS) operation related Structures, Systems and Components (SSCs) in the UK HPR1000. Chapter 16 provides the general description of the BFX building, which is one of the buildings included in the SFIS

Chapter	Interface
	operations.
Chapter 30 Commissioning	Chapter 30 presents the arrangements and requirements for commissioning. Chapter 16 provides the containment system design requirements which align with the requirements described in Chapter 30.
Chapter 31 Operational Management	Chapter 31 provides the EMIT arrangements to manage ageing and degradation effects. Chapter 16 presents the in-service inspection and testing for the reinforced concrete structures of buildings.
Chapter 33 ALARP Evaluation	Chapter 33 presents the overall ALARP demonstration. Chapter 16 provides the ALARP demonstration for the civil structures according to the principles described in Chapter 33.

Note: this Chapter has implemented the requirements management coding system for BFX, internal containment and the common raft foundation. The requirements management coding system will be applied to all civil engineering buildings and structures in the site licensing phase.

16.3 Safety Functional Requirements and Engineering Requirements

16.3.1 Safety Functional Requirements

The decomposition of safety functions (Reference [3]) provides a systematic and comprehensive list of safety functions for the UK HPR1000, which is applied for the entire Nuclear Power Plant (NPP) and all relevant SSCs. The three fundamental safety functions and the extra safety function, which are referred as “R - Reactivity control”, “H - Heat removal”, “C – Confinement” and “E – Extra”, are top-level safety functions. The other levels of safety functions are decomposed from the top-level safety functions.

The LLSFs relevant to Civil Engineering are screened and identified. The detailed screen of the LLSFs is provided in Appendix 2 “Screening of the LLSFs relevant to Civil Engineering”.

Then, safety functional requirements for civil structures are then derived: housing, support and protection for SSCs, and confinement of radioactive materials as well as

limitation of accidental radioactive releases.

These safety functional requirements have been further decomposed into the following safety functions.

a) F1: Function to provide structural support to SSCs;

b) F2: Function to provide protection to SSCs against external hazards;

c) F3: Function to provide protection to SSCs against internal hazards;

d) F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release;

e) F5: Function to maintain specific internal building environments appropriate for SSCs during normal operating and accident situations;

f) F6: Function to satisfy the requirements of decommissioning which include supporting SSCs, providing a barrier and confining the radioactive materials.

The relationship between the LLSFs relevant to the civil structures and the safety functions of civil structures is shown in table T-16.3-1.

T-16.3-1 Relationship between LLSFs and Safety Functions

LLSFs related to the Civil Structures	**Safety Functions of Civil Structures**
H4-2.Protect the fuel pool from leakage.	F4
C3-2. Maintain containment building structural integrity	F1
C3-3 Prevent leaks from containment building	F4
C5-2.Store the radioactive material as liquid wastes	F4
C5-3.Store the radioactive material as solid wastes	F4
C6-1.Shield against radiation	F4
C6-3 Limit accidental radioactive releases	F4
E1-4.Provide other supporting for the performance of safety functions	F1
E2-3.Protect internal hazards risk	F3
E2-4.Protect external hazards risk	F2

Note: F5 and F6 are provided by other design considerations. F5 is provided by heating, ventilation and air conditioning (HVAC) systems maintaining the correctly controlled environment and atmosphere appropriate for SSCs inside the buildings. F6 is provided as described in Chapter 24.

Detailed information about the safety functional requirements for internal and external

containment, BRX internal structures, BSA/BSB/BSC, BFX, BNX, BWX, BDA/BDB/BDC/BDU/BUV, BEJ, BPX and BEX are shown in section 4 of the basis of safety case reports for the corresponding buildings, Reference [4], [5], [6], [7], [8], [9], [10], [11], [12] and [13] respectively.

16.3.2 Classification and Categorisation

The safety category and safety classification are identified based on the safety function of the structures. According to the principles for the classification of SSCs developed in Chapter 4 Sub-chapter 4.4.5, structures are classified by the highest classification and category of components housed in the building. The functional classifications of different structures are F-SC1, F-SC2 and F-SC3 separately.

The detailed process for the classification and categorisation of structures is described in section 5 of the basis of safety case reports, References [4], [5], [6], [7], [8], [9], [10], [11], [12] and [13].

The classification and category of structures is listed in Table T-16.3-2.

T-16.3-2 Classification and Category of Structures

Building	Function Categorisation	Functional Classification	Seismic Category
BRX	FC1	F-SC1	SSE1
BSA/BSB/BSC	FC1	F-SC1	SSE1
BFX	FC1	F-SC1	SSE1
BNX	FC3	F-SC3	SSE1
BDA/BDB/BDC/ BDU/BDV	FC1	F-SC1	SSE1
BWX	FC3	F-SC3	SSE1
BEJ	FC3	F-SC3	SSE1
BEX	NC	NC	SSE2
BPX	NC	NC	SSE2

a) FC1: Category 1, FC3: Category 3, NC: Non-Categorised.

b) F-SC1: Functional Class 1, F-SC3: Functional Class 3, NC: Non-Classified.

c) SSE1: Seismic Category 1, SSE2: Seismic Category 2.

d) Function categorisation and functional classification of BDU/BDV are FC3 and F-SC3 respectively. Considering BDU is integrated with BDA/BDB and BDV is integrated with BDB, seismic category of BDU/BDV is same as BDA/BDB/BDC, i.e., SSE1.

16.3.3 Engineering Requirements

The engineering requirements of civil structures are identified to ensure that the safety functional requirements can be achieved. The engineering requirements comprise the structural behaviour requirements and other design considerations.

16.3.3.1 Structural Behaviour Requirements

In order to meet the safety functional requirements F1-F4, several structural behaviour requirements are identified based on the safety functional requirements and classification and categorisation of the structures. These structural behaviour requirements are ensured in design by meeting the relevant design criteria provided in the codes and standards.

The structural behaviour requirements include strength, stability, durability, serviceability, water tightness and air tightness.

a) D1: Strength---Structures are strong and robust. Structures and structural members are designed to have required strength at all sections;

b) D2: Stability---Structures cannot collapse, and structures can maintain overall structural stability;

c) D3: Durability---Structures can satisfy the design life requirement;

d) D4: Serviceability---The deformation of the structure and structural members are limited and compatible with the requirements of the systems and components they contain;

e) D5: Water tightness---Structures are required to retain or exclude liquids;

f) D6: Air tightness---Structures are required to contain gases under pressure with a limited rate of gas leaks.

For the BRX internal containment and other nuclear structures, the structural behaviour requirements are presented as follows.

a) BRX internal containment

   For design basis conditions, the structures shall be designed to ensure that the Strength (D1) and Stability (D2) requirements can be achieved according to Reference [14]. For D1 requirements, the stresses of structures and structural members shall be designed to be less than the allowable stresses. For D2 requirements, the global structure cannot collapse. Structures shall maintain overall structural stability to prevent overturning, sliding and flotation.

   Complying with the provisions specified in Reference [14] and [15] regarding durability and serviceability, the structures shall satisfy Durability (D3) and Serviceability (D4) behaviour requirements. For D3, the structures shall be

designed to ensure that any deterioration over its design working life shall not impair the performance of the structure. For D4, the deformation of the structure and structural members shall be limited and compatible with the requirements of the equipment according to the relevant design codes.

Air tightness (D6) for the internal containment shall be achieved by two barriers. The first barrier is a steel liner and the second barrier is a prestressed concrete structure. The second barrier achieves airtightness by controlling the concrete stress to limit the development of cracks and by adding structural inspection measures.

For beyond design basis conditions, the structural evaluations include the ultimate capacity evaluation for internal containment and cliff-edge effects from extreme environmental hazards for civil engineering structures.

b) Other nuclear structures

For design basis conditions, the other nuclear structures shall be designed to ensure that the Strength (D1) and Stability (D2) requirements can be achieved according to Reference [16]. For D1 requirements, the design strength of the structures and the structural members shall be designed to be at least equal to the required strengths at all sections. For D2 requirements, the global structure cannot collapse. Structures shall maintain overall structural stability to prevent the overturning, sliding and flotation.

Complying with the provisions specified in Reference [16] and [15] about durability and serviceability, the structures shall satisfy Durability (D3) and Serviceability (D4) behaviour requirements. For D3, the structures shall be designed to ensure that any deterioration over its design working life shall not impair the performance of the structure. For D4, the deformation of the structure and structural members shall be limited and compatible with the displacement requirements according to the relevant design codes.

The first barrier for water-tightness (D5) shall be achieved by an internal stainless steel liner. The most important pools with steel liners are equipped with a leakage detection system. The concrete structural members are used as the second barrier by controlling concrete crack widths.

Air tightness (D6) for the external containment shall be achieved by the thickness of walls and the air-tightness of penetrations. The crack width limitation for the external containment is not required specifically for the air-tightness requirement. The ventilation system placed in the annulus between the external containment and internal containment also ensures the air-tightness of the external containment.

For beyond design basis conditions, the structural evaluations include the

structural evaluation for aircraft impact and cliff-edge effects from extreme environmental hazards.

The overall safety functional requirement schedules for the BFX and the internal and external containments are documented in *Civil Engineering Schedule Report*, Reference [17].

16.3.3.2 Other Design Consideration

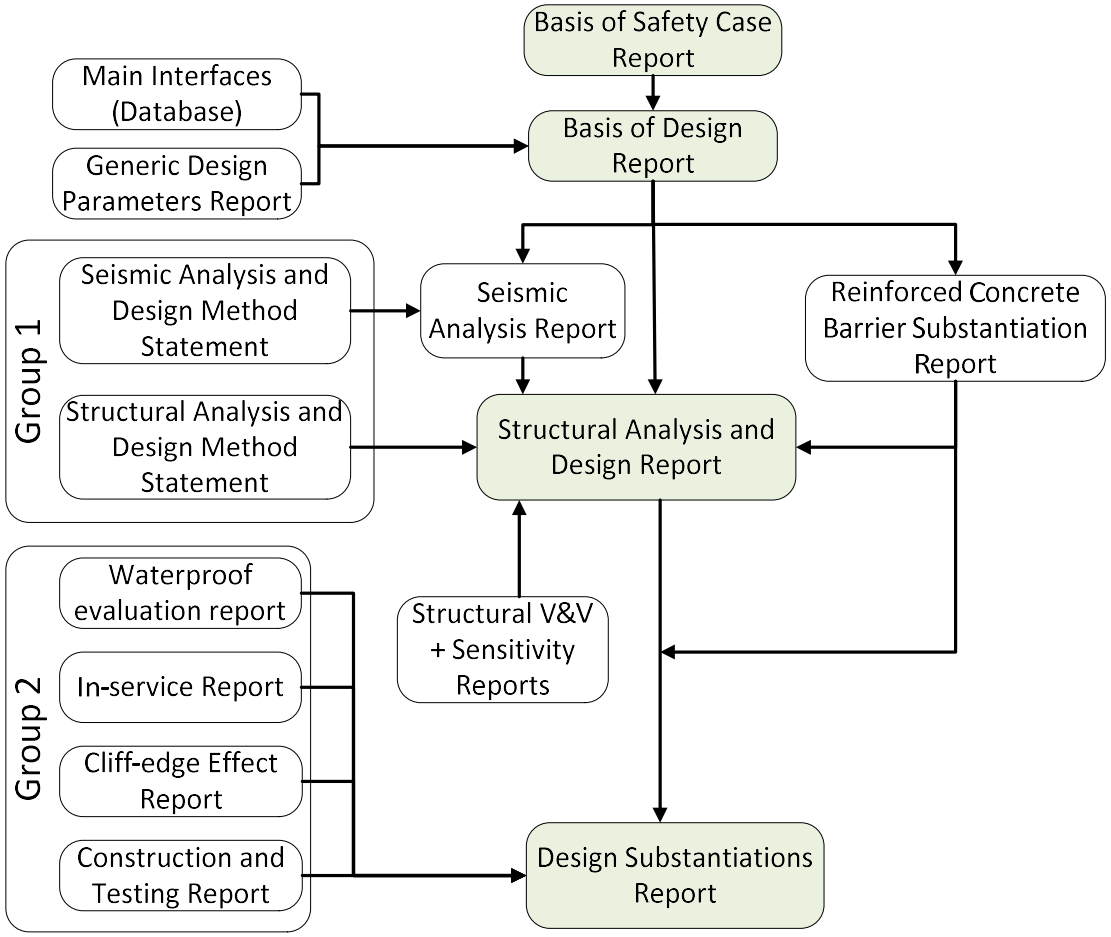
Other design considerations have been made in addition to the structural behaviour requirements to satisfy safety functions F5 and F6.

For safety functional requirement F5, the specific internal building environment can be maintained by HVAC systems. For safety function F6, structural requirements are derived from the plant decommissioning requirements which are described in the Chapter 24 Sub-chapter 24.4 and are integrated into the design of buildings.

16.3.3.3 Requirements Management

The requirements management process, including the coding system, is presented in the Chapter 20. This Chapter has implemented the requirements management coding system for BFX, internal containment and common raft foundation.

For each building within GDA scope, a series of documents are set as the safety case for the building, which include the basis of safety case report, the basis of design report, structural analysis and design report, and design substantiation report.



F-16.3-1 Delivery of Design Requirements in Documents

The basis of safety case report is the entry document for the functional requirements as it contains the civil engineering schedule.

The civil engineering schedule in the basis of safety case report is composed of related safety functions, design conditions/hazards, upstream reference, safety function requirement/hazard protection requirement (coding), engineering requirements and acceptance criteria. Safety functional requirements are specified for the overall structure and for typical structural members in the buildings. The typical structural members include internal and external walls/slabs, pools and raft foundations which serve as barriers and non-barriers against hazards.

The design substantiation report provides the substantiation of the civil engineering structure ("exit" document). The design substantiation report draws together all safety functional requirements (include coding) from the basis of safety case report and provides design substantiation based on evidence provided in other reports, such as the basis of design report, structural analysis report, seismic calculation report etc.

16.4 Applicable Codes and Standards

The applicable codes and standards are selected based on the selection principles

determined in the Chapter 4.4.7. During the UK HPR1000 design, UK regulations need to be followed, which include *Construction (Design and Management) (CDM) Regulations*, Reference [18], and *Building Regulations*, Reference [19]. The codes and standards which are recognised as Relevant Good Practice (RGP) are considered.

In order to select applicable codes and standards, an approach to codes and standards for civil structures which reflects the ALARP principle is carried out. More details about the approach taken to select codes and standards are described in Reference [20].

The strategy for the application of codes and standards for the UK HPR1000 considers the following:

a) Adoption of American codes and standards for strength design.

b) Adoption of a combination of American and European codes and standards for serviceability design.

c) Material specification is in accordance with European and British codes and standards.

The compliance and suitability analysis of codes and standards in civil engineering has been carried out. The application of US nuclear design codes and standards in the UK has been reviewed to identify the gaps between US codes and UK context and to provide an approach/methodology to eliminate these gaps. More details about the application of codes and standards for the UK HPR1000 are given in Reference [21].

The codes and standards listed below are either the latest version at the time of production of this report or the most compatible to the primary code or standard. The primary codes and standards are supported by a series of supplementary documents which are listed in their relevant sections or reports.

16.4.1 Seismic Analysis

a) ASCE/SEI 4-16, Seismic Analysis of Safety-Related Nuclear Structures, American Society of Civil Engineers, 2017, Reference [22];

b) ASCE/SEI 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, American Society of Civil Engineers, 2005, Reference [23].

16.4.2 Structural Analysis and Design

a) Internal Containment Structure Design

ASME Boiler & Pressure Vessel Code, Section III, Division 2, Subsection CC: Code for Concrete Containments, American Society of Mechanical Engineers, July 2017, Reference [14].

b) Other Nuclear Safety-Related Structure Design

1) ACI 349M-13, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, American Concrete Institute, August 2013, Reference [16];

2) BS EN 1992, Eurocode 2, Design of Concrete Structures and its UK National Annex, December 2004, Reference [15];

3) ETC-C, EPR Technical Code for Civil Works, ETC-C 2010 Edition, Reference [24].

c) Foundation Design

1) BS EN 1997, Eurocode 7, Geotechnical Design and its UK National Annex, BS EN 1997, February 2009, Reference [25];

2) ACI 349M-13, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, American Concrete Institute, August 2013, Reference [16];

3) NUREG-0800, US NRC Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants-LWR Edition. Section 3.8.5: Foundations, Revision4, September 2013, Reference [26].

d) Aircraft Impact Analysis

1) IAEA, Safety Aspects of Nuclear Power Plants in Human Induced External Events: Assessment of Structures, IAEA Safety Report Series No.87, 2018, Reference [48].

2) ERIN Engineering & Research, Inc. Methodology for Performing Aircraft Impact Assessments for New Plant Designs, NEI 07-13, 2011, Reference [49].

16.4.3 Materials

a) BS EN 206: 2013, Concrete - Specification, Performance, Production and Conformity, Reference [27];

b) BS EN 10080:2005, Steel for the Reinforcement of Concrete - Weldable Reinforcing Steel –General, December 2005, Reference [28];

c) BS 4449:2005, Specification for Carbon Steel Bars for the Reinforcement of Concrete, November 2005, Reference [29];

d) Pr EN 10138:2009, Prestressing Steels, August 2009, Reference [30].

16.4.4 Metrication

The design and construction of the UK HPR1000 is undertaken based on the

international metric system. However, if the reference codes and standards are only available based on the imperial system, there are two available options for applying the UK HPR1000 design:

a) Conversion of the codes and standards to the international metric system for carrying out the UK HPR1000 design;

b) Conversion of the design input from metric to imperial units, undertake the design using imperial units, and finally convert the design output from imperial to metric units.

16.5 Seismic Analysis

This Sub-chapter introduces the seismic analysis of structures, which includes seismic input, modelling of structures and seismic analysis of SSE1 structures, SSE2 structures and non-nuclear seismic structures.

The seismic categorisations of different structures are SSE1, SSE2 and non-nuclear seismic structures. The SSE1 categorisation buildings are designed to remain elastic under design basis earthquake, the SSE2 classification buildings are designed to not affect the function of nuclear safety-related items and the non-nuclear seismic structures are designed in line with normal industrial codes and standards.

16.5.1 Seismic Input for SSE1 and SSE2 Structures

16.5.1.1 Design Response Spectra

For the GDA phase, the European Utility Requirement (EUR) spectra, Reference [31], is adopted as the Design Basis Earthquake (DBE) input motion for the seismic analysis of SSE1 and SSE2 structures. The horizontal and vertical peak ground accelerations are 0.30 g and 0.20 g respectively. The design seismic excitation is applied at the bottom level of the foundation.

In the site specific design phase, site specific design response spectra will be developed and used for the seismic analysis.

16.5.1.2 Design Time Histories

The soft and medium soil conditions which are divided into three subgrade conditions are assumed in GDA. For each subgrade condition, a single set of time histories consisting of two horizontal components and one vertical component are generated to perform time history analyses of civil structures.

The set of time histories is compatible with the design response spectra and satisfies the requirements defined in Reference [23], including the requirements about time increment and duration, amplitude, directional correlation and power spectral densities. The method used to develop time histories refers to Appendix A of the report *Generic Design Parameters for Civil Engineering*, Reference [32].

16.5.1.3 Subgrade Properties

Three kinds of subgrade conditions corresponding to the soft and medium soil conditions are assumed in GDA. The generic subgrade properties are defined in the section 3.4 of the report *Generic Design Parameters for Civil Engineering*, Reference [32].

16.5.2 Modelling of Structures

The seismic response of a nuclear structure is determined by a mathematical model of the structure. The structures are simulated using three-dimensional finite element models that can represent the properties of mass and stiffness appropriately. The mass of the structure includes self-weight, equipment, distribution systems, effective live load, etc. The hydrodynamic effects in the global seismic analysis are considered as added mass of the model. The stiffness of the structural component is represented based on the material properties and the geometric features.

The types of finite elements required are determined by the type of the structure and the response parameter of interest. The structural walls and slabs are modelled using shell elements, and the columns and beams are modelled using beam elements. A reasonable element mesh size of elements is adopted to achieve results of an acceptable accuracy, taking into account the different analysis purposes of the finite element models.

The methods used for the detailed modelling of structures are described in *Overall Seismic Analysis Method Statement*, Reference [33].

16.5.3 Seismic Analysis of SSE1 Structures

All SSE1 structures within the UK HPR1000 are designed to be Limit State D as per ASCE/SEI 43-05, Reference [23], and will exhibit essentially elastic behaviour with negligible damage during and after a DBE. The seismic analysis considering Soil-Structure Interaction (SSI) analysis is undertaken in the ACS SASSI environment and the seismic structural stresses are calculated using ANSYS finite element software.

For the seismic analysis of the SSE1 structures on the common raft foundation, a single seismic analysis is undertaken encompassing all of the buildings located on the common raft. For the buildings not on the common raft and with their individual raft foundations, the seismic analysis is undertaken on a building by building basis.

The main contents of seismic analysis of SSE1 structures include the Floor Response Spectra (FRS), seismic member demands, seismic gaps checking, overall stability of the structures, Verification and Validation (V&V) and sensitivity studies.

16.5.3.1 Development of Floor Response Spectra

The FRS are developed by using the acceleration time series which are determined

from the SSI analysis in the ACS SASSI environment. For each subgrade condition, the SSI analysis is conducted and the FRS is generated for each floor of the building, and then the FRS of each subgrade is enveloped to get the generic design FRS.

To account for the uncertainties in the dynamic behaviour of the soil and structures, the approach of peak broadening is adopted.

The procedure of developing the FRS is described in *Overall Seismic Analysis Method Statement*, Reference [33].

16.5.3.2 Development of Seismic Member Demands

The overall seismic response of structures, including nodal displacement histories for different generic subgrade conditions are calculated in the ACS SASSI environment. Nodal displacements histories for each of the three directional earthquake components are extracted from ACS SASSI and co-directional responses to each of the three earthquake components are combined using algebraic sums at each time step in the time domain according to ASCE 4-16. The combined nodal displacement histories for all degrees of freedom are applying simultaneously as input boundary constraints in the time domain to the ANSYS seismic model to perform the subsequent analysis time step by time step and the maximum member demands are recorded for design.

The detailed information about calculations of seismic member demands is described in *Overall Seismic Analysis Method Statement*, Reference [33].

16.5.3.3 Seismic Gaps Evaluation

The design requirement for seismic gaps used in UK HPR1000 conforms to Section 7.3 of ASCE/SEI 43-05, Reference [23]. The minimum separation maintained between adjacent buildings is twice the Square Root Sum of the Squares of D1 and D2, where D1 and D2 are the maximum elastically computed displacements, relative to the base of the foundation of each building.

16.5.3.4 Overall Stability

The methodology for the overall stability of buildings is described in Sub-chapter 16.6.4.3.

16.5.3.5 Verification and Validation for Seismic Analysis

Within GDA, the V&V and sensitivity studies for the various ANSYS and ACS SASSI analytical models of the UK HPR1000 and the structural design process are performed, including the validation of the models, mesh size, verification of calculation process, results transfer between different models, sensitivity studies of damping values, stiffness, time histories, embedment effects, and structure-soil-structure interaction.

The V&V ensure that the analytical models and subsequent results give an accurate

representation of the UK HPR1000 structural behaviour. The sensitivity studies assess the influence of the analysis parameters/approaches on the results and demonstrate that the chosen parameters/approaches are sufficiently appropriate based on the conclusions from the sensitivity studies.

16.5.4 Seismic Analysis of SSE2 Structures

SSE2 structures are designed to ensure that the DBE action could not cause an unacceptable impact on adjacent SSCs of SSE1 structures. The seismic gaps between the SSE2 structures and adjacent SSE1 structures are checked to ensure that there is no interaction between these structures. SSE2 structures are designed to be Limit State C as per the requirement of ASCE/SEI 43-05, Reference [23].

The detailed seismic analysis method of SSE2 structures is described in *Overall Seismic Analysis Method Statement*, Reference [33].

16.5.5 Analysis of Non-Nuclear Seismic Structures

Non-nuclear seismic structures are outside the scope of GDA. The design of these structures adopts the normal industrial codes and standards and will be developed in the site specific stage.

16.6 Design of Nuclear Safety-Related Structures

This Sub-chapter introduces the design of nuclear safety-related structures, which includes structural description, design principles, loads and load combinations, methods of analysis and design, and substantiation.

This Sub-chapter provides the description of the safety classified structures which include BRX, BSA/BAB/BSC, BFX, BNX, BDA/BDB/BDC/BDU/BDV, BWX and BEJ. The BEX and BPX are non-classified structures and the descriptions are provided in the basis of safety case reports, Reference [13] and [12].

16.6.1 Structural Description

16.6.1.1 Nuclear Safety-Related Structures Layout

The buildings on the common raft foundation include BRX, BFX and the Safeguard Buildings. BNX, BWX, BEJ, BPX, BEX, BDA/BDC/BDU and BDB/BDV are separately located on independent raft foundations. The common raft buildings are separated from the adjacent buildings by a 200 mm gap.

The structural relationship of buildings on the common raft foundation is described as follows:

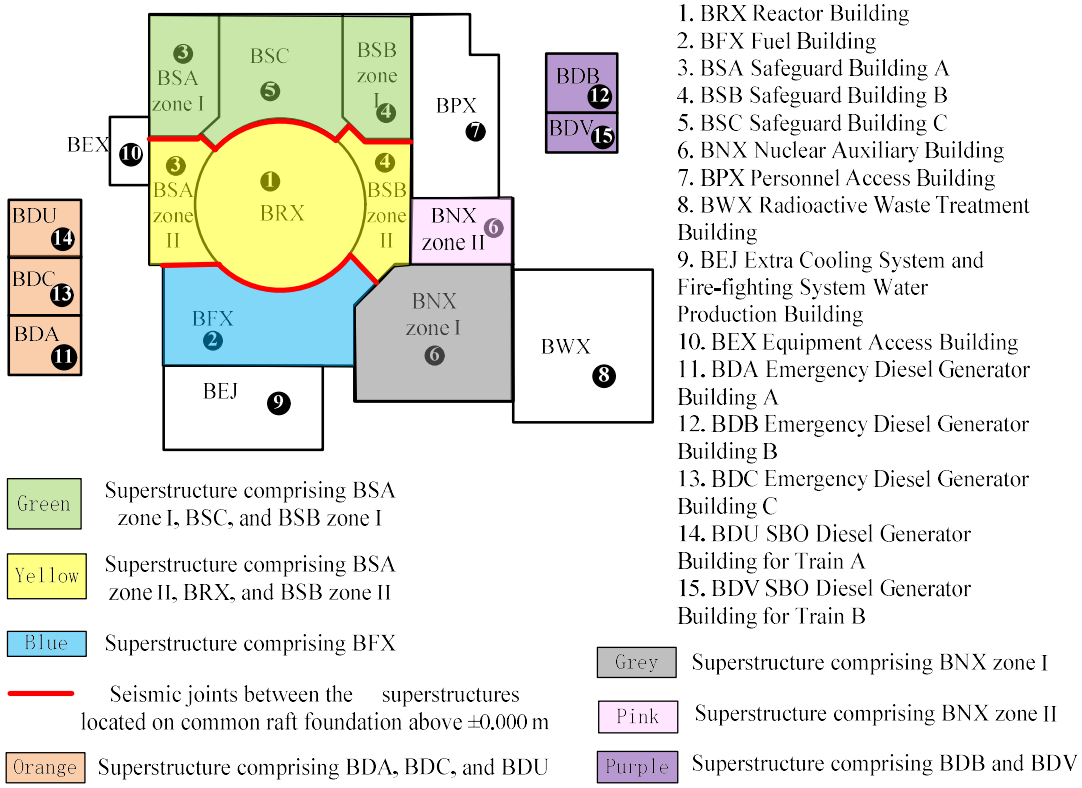
a) External Containment, BFX and Safeguard Buildings are connected together below ±0.00 m elevation;

b) BSA-II, External Containment and BSB-II are connected together above ±0.00 m

elevation;

c) BFX, BSA-II + External Containment + BSB-II and BSA-I + BSC + BSB-I are separated from each other by 100 mm structural joints above ±0.00 m elevation.

The layout of nuclear safety-related structures is shown in F-16.6-1. Detailed layout information of these structures is provided in *Overview of the UK HPR1000 Civil Structures*, Reference [34].



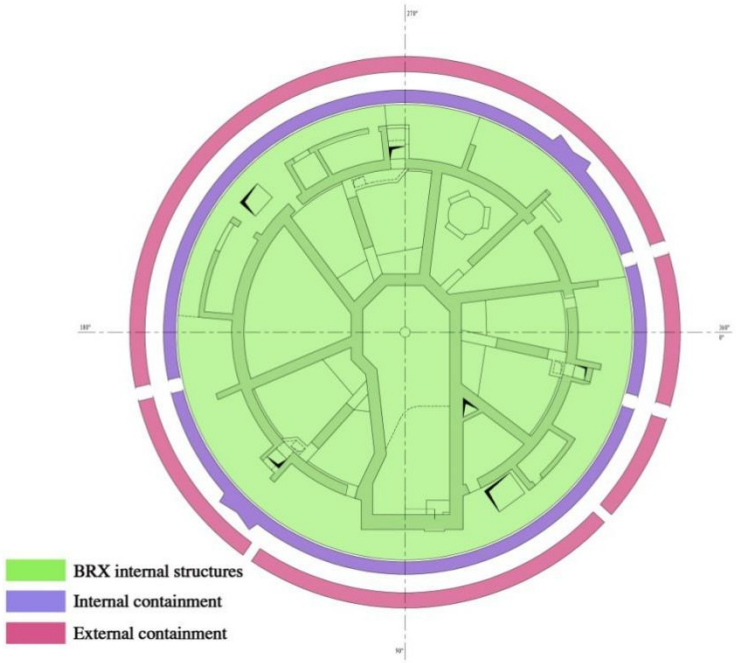
F-16.6-1 Nuclear Safety-Related Structures Layout

16.6.1.2 Reactor Building

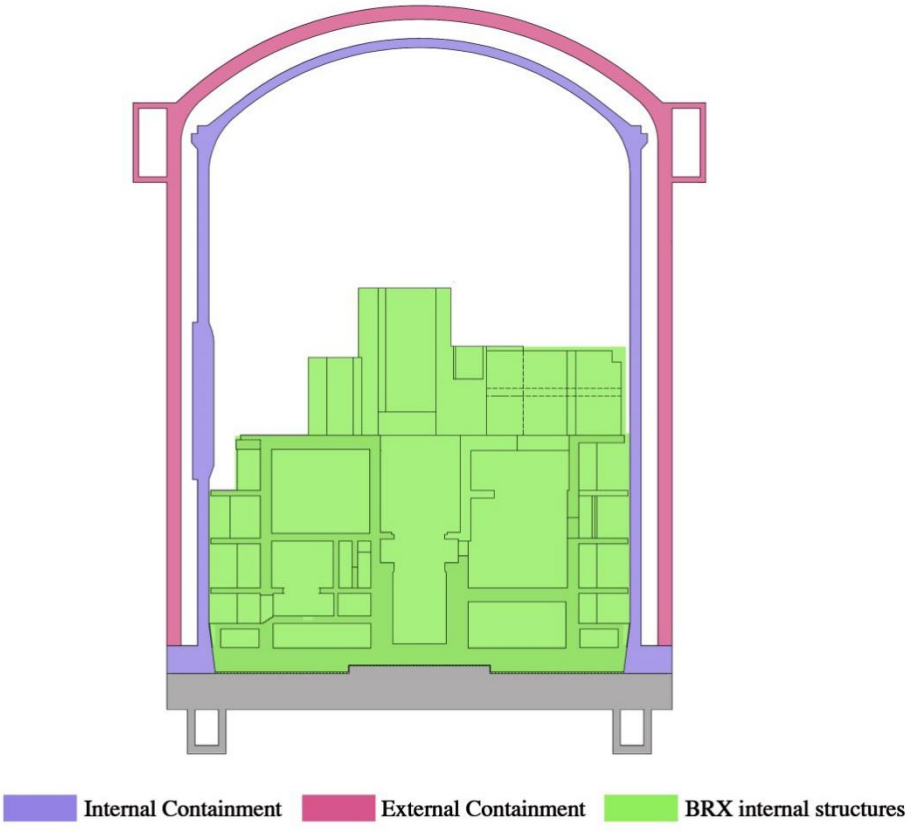
The reactor building consists of the internal containment, external containment and internal structures. It is located in the centre of the common raft foundation.

The external containment is located outside of the internal containment with an annulus set between them. The internal structures are housed within the internal containment.

The typical plan view and cross section of the reactor building are shown in F-16.6-2 and F-16.6-3 respectively.



F-16.6-2 Typical Plan View of the Reactor Building



F-16.6-3 Typical Section View of the Reactor Building

a) Internal Containment

The internal containment is a prestressed concrete structure, the inner surface of

which is covered with a steel liner. It is comprised of a cylindrical wall, dome, ribs and ring belt. Two vertical ribs are located on the cylindrical wall symmetrically. A prestressing gallery is set below the common raft foundation for the tensioning of prestressed tendons.

The main geometry of the internal containment is as follows:

1) The internal diameter is ~45m.

2) The height is ~68m.

3) The thicknesses of the cylindrical wall and dome are 1.2m and 1.0m respectively.

4) The elevation of the equipment hatch centre is +20.800m.

5) The elevation of ring belt is between +45.000m and +51.000m.

The typical plan view of components in the reactor building is shown in F-16.6-4 and the typical cross section of components in the reactor building is shown in F-16.6-5.

The internal containment prestressing systems uses bonded tendons, which include horizontal tendons and vertical tendons. The horizontal tendons surround the cylinder wall and are anchored to the ribs. The vertical tendons are anchored to the bottom of the raft foundation at one end and anchored to the ring belt at the other end.

The steel liner doesn't have any role for structural resistance and is composed of the anchoring system and the liner plate. The anchoring system is anchored to the inner concrete surface of the internal containment and at the top of the reactor building foundation. The anchoring system of the steel liner is buried within the concrete structure and the steel liner plate is welded to the anchoring system. The anchoring system is composed of grids at right angles. The anchoring system is used to reinforce the steel liner and ensure its stability during construction and operation periods. The steel liner can also serve as the construction formwork of the internal containment wall.

The monitoring system is embedded in the internal containment, which will monitor the structural behaviour of the internal containment throughout the service lifetime. More detailed information is described in Sub-chapter 16.9.

In order to allow for the passage of equipment, personnel and pipes, the internal containment and external containment are equipped with an equipment hatch, two personnel access airlocks and other small penetrations. There are also some typical secondary structural members arranged in the internal containment, for example, the fuel transfer tube and the polar crane bracket.

b) External Containment

The external containment is located outside of the internal containment. It is a reinforced concrete structure composed of a cylindrical wall and dome. A passive water tank is located on the top of the cylindrical wall.

The main geometrical parameters of the external containment are the following:

1) The internal diameter is ~51m;

2) The height is ~70m;

3) The thickness of the cylindrical wall and dome is 1.5m.

The typical plan view of components in the reactor building is shown in F-16.6-4 and the typical cross section of components in the reactor building is shown in F-16.6-5.

c) BRX Internal Structures

The BRX internal structures are located in the centre of the BRX. The plan view of the internal structure is a circle with a radius of 22.5m. The total height is ~41m and the height above the ground is 33.2m.

The BRX internal structures comprise three main functional levels below:

1) +1.200m (main equipment floor).

2) +5.660/+7.160m (reactor pool).

3) +17.500m (service floor).

4) Other auxiliary levels (served by stairs and a lift).

The typical plan view of components in the reactor building is shown in F-16.6-4 and the typical cross section of components in the reactor building is shown in F-16.6-5.

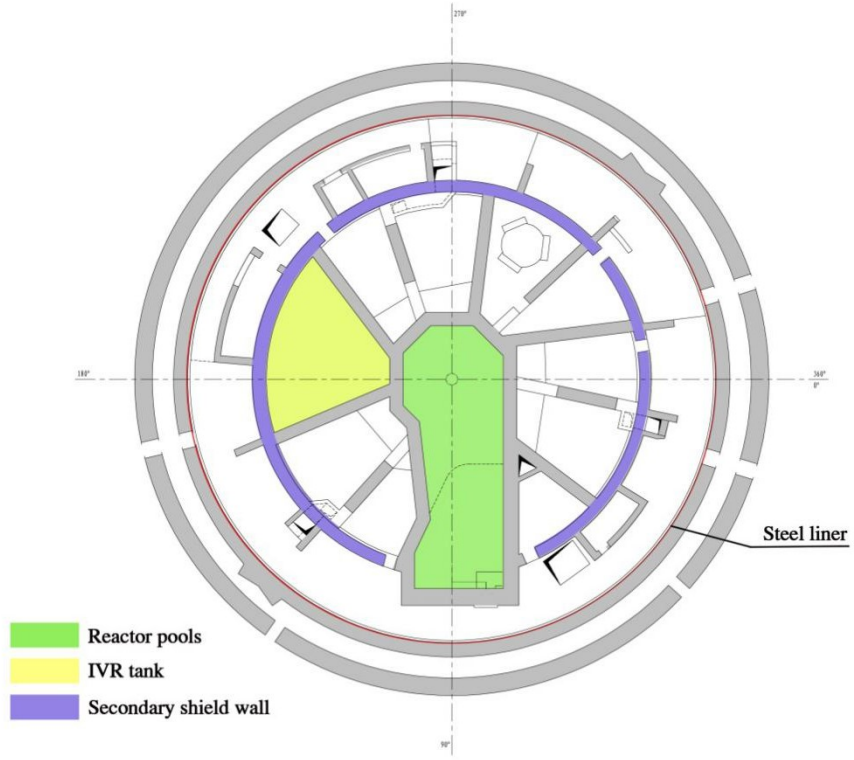
The BRX internal structures include the reactor cavity, secondary shield wall and reactor pools.

The reactor cavity is composed of a cylinder which is coaxial with the reactor building. The cavity supports the reactor pressure vessel and acts as radiation shielding to protect personnel and equipment. The secondary shield walls consist of walls, support slabs and floors. The walls and the reactor cavity form the compartments that enclose all the main reactor cooling systems and equipment.

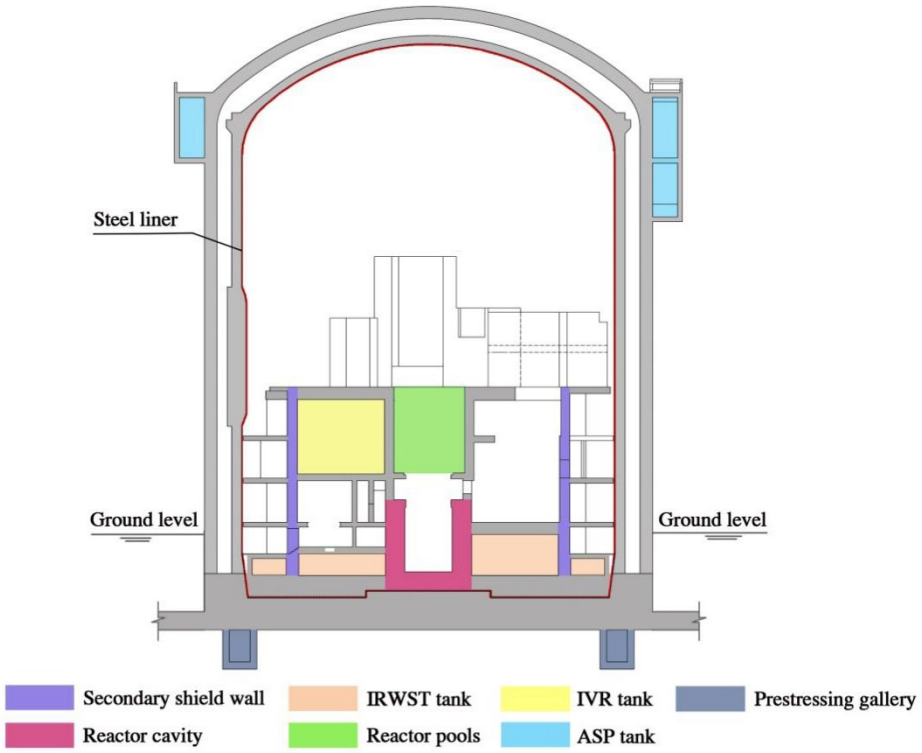
The reactor pool is composed of two parts, the cavity pool and the reactor internal component pool. The pool is lined internally with a stainless steel liner. The liner forms the primary barrier to potential leakage and the concrete walls and slabs form the secondary barrier.

The BRX internal structures are reinforced concrete walls and slab structures. The horizontal loading is transferred by the slabs to the walls. The vertical loading is supported by the slabs and is transferred by walls of the secondary shield wall, reactor cavity and reactor pools to the common raft foundation.

Detailed design requirements and the layout of the BRX internal structures are provided in *Basis of Safety Case for BRX Internal Structures*, Reference [5].



F-16.6-4 Typical Plan View of Components in the Reactor Building



F-16.6-5 Typical Cross Section of Components in the Reactor Building

16.6.1.3 Other Buildings

a) Safeguard Buildings

The safeguard buildings which include the BSA, BSB and BSC are located around the BRX. The main geometry of the safeguard buildings is as follows:

1) The length and width of the safeguard buildings are ~78.2m and 81.4m respectively;

2) The length, width and height of the BSA are ~78.2m, 20.6m and 39.6m respectively (including the raft foundation), which includes two floors below ±0.000m and six floors above ±0.000m. The BSB has approximately the same dimensions as the BSA;

3) The length, width and height of the BSC are ~40.2m, 38.45m and 43.4m respectively (including the raft foundation), which includes two floors below ±0.000m and seven floors above ±0.000m.

The typical plan view of the safeguard buildings is shown in F-16.6-6. The typical cross section of the safeguard buildings is shown in F-16.6-7.

The major functional areas housed by the safeguard buildings are as follows:

1) Main control room;

2) Main stream and feed water piping area;

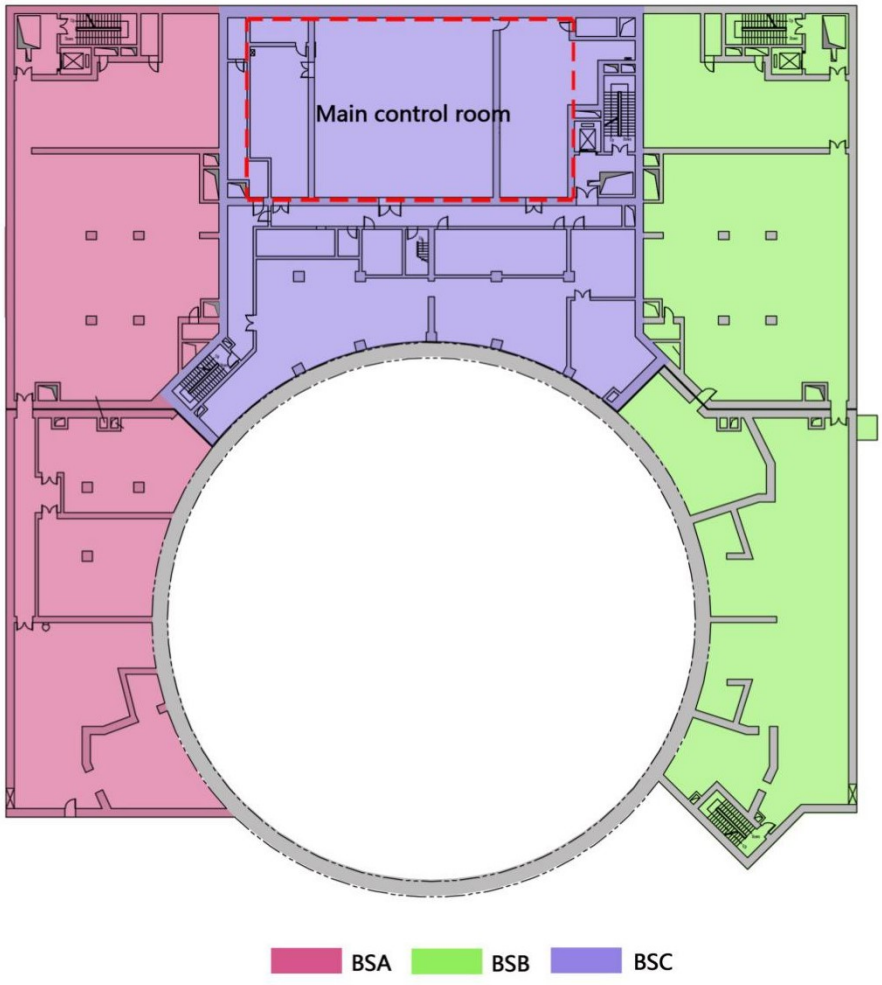
3) Safety-related mechanical and electrical systems area;

4) Instrumentation & Control (I&C) systems area.

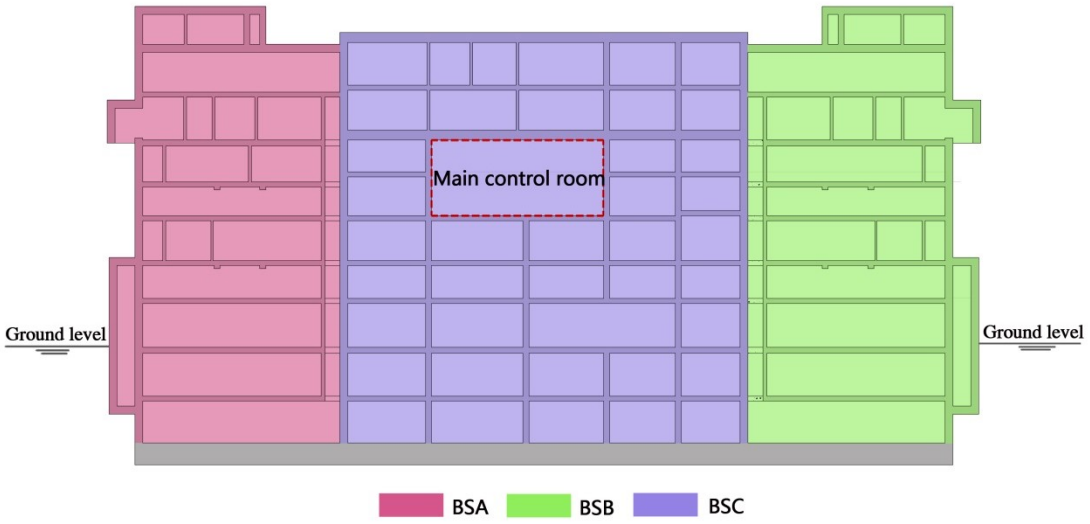
The safeguard buildings are a reinforced concrete structure consisting of walls, slabs, some reinforced columns and beams in local areas. The thickness of the walls is in the range of 0.6m to 1.0m. The thickness of slabs is in the range of 0.5m to 1.0m.

The horizontal loading is transferred by the roofs and slabs to the walls and columns. The vertical loading is transferred by the walls and columns to the common raft. The reinforced concrete walls on the common raft provide stiffness for the common raft and transfer the final loading to the common raft foundation.

Detailed design requirements and the layout of the safeguard buildings are provided in *Basis of Safety Case for Safeguard Buildings*, Reference [6].



F-16.6-6 Typical Plan View of the Safeguard Buildings



F-16.6-7 Typical Cross Section of the Safeguard Buildings

b) BFX

The BFX has ten main levels, with eight floors above ±0.000m and two floors below ±0.000m. The main geometry of the BFX is as follows:

1) The length of the BFX is ~60.6m;

2) The width of the BFX is ~36.95m;

3) The height of the BFX is ~42.5m.

The typical plan view of the BFX is shown in F-16.6-8. The typical cross section of the BFX is shown in F-16.6-9.

The major functional areas housed by the BFX are listed below:

1) The mechanical equipment area;

2) Preliminary mechanical component area;

3) The area for ventilation and air conditioning equipment.

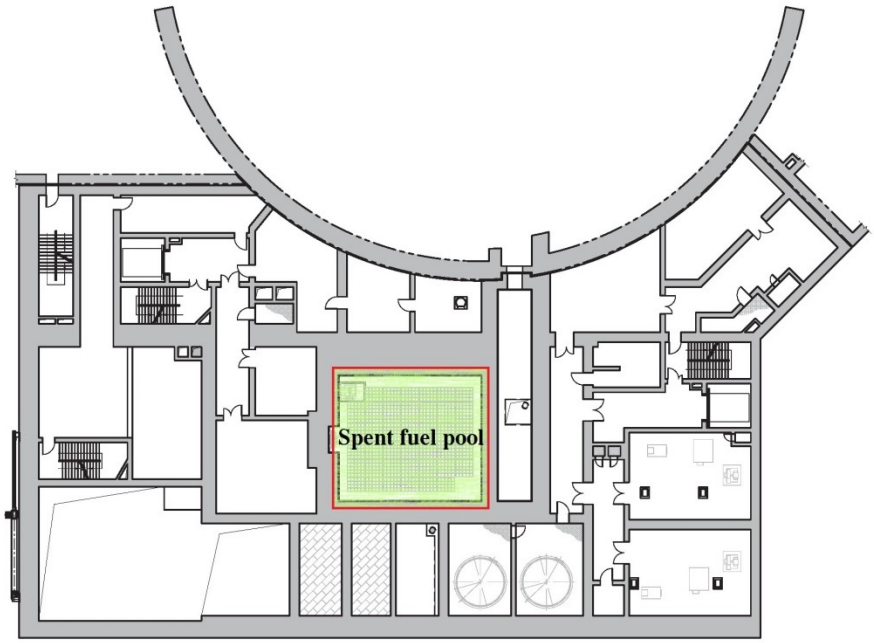
The BFX is a reinforced concrete structure consisting of walls and slabs. The thicknesses of walls range from 0.3m to 1.8m. The thicknesses of slabs range from 0.4m to 0.7m. The thickness of the roof is 1.0m.

The spent fuel pool is a reinforced concrete structure which is lined with stainless steel liner to ensure leak-tightness. The double barriers for Water Tightness (D5) are achieved by the stainless steel liner and concrete wall/slab. The steel liner is equipped with a leakage detection system. The liner is anchored to the concrete walls of pool and has no structural function. The pool is located at the middle of the structure and the thickness of the walls and floor are at least 1.8m. The walls

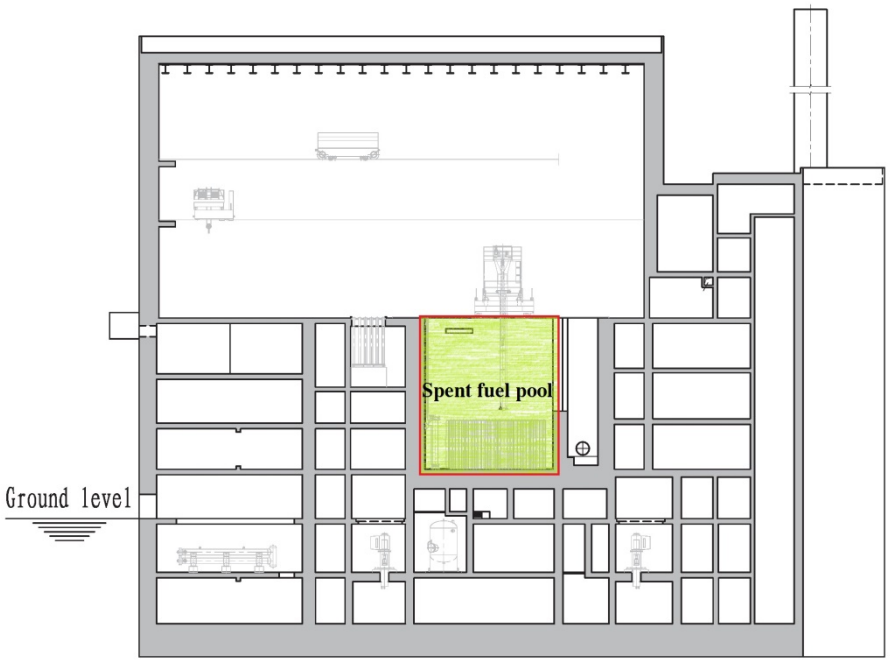
of the spent fuel pool also act as the structure bearing wall. The thickness of the walls is designed to meet the requirements for radiation shielding and structure bearing.

The horizontal loading is transferred by the roofs and slabs to the walls. The vertical loading is transferred by walls to the common raft foundation. The reinforced concrete walls on the common raft foundation provide stiffness for the raft foundation and transfer the final loading to the raft foundation.

Detailed design requirements and the layout of the BFX are provided in *Basis of Safety Case for BFX*, Reference [7].



F-16.6-8 Typical Plan View of the BFX



F-16.6-9 Typical Cross Section of the BFX

c) BNX

The raft foundation and superstructure of the BNX are divided into zone I and zone II, which are isolated by a structural gap with a 200mm width. The main geometry of the BNX is as follows:

1) The length and width of BNX zone I are ~43.2m and 50.2m respectively, including three floors below ±0.000m and seven floors above ±0.000m. The height of the superstructure of zone I is ~31m;

2) The length and width of BNX zone II are ~20.8m and 32.1m respectively, including two floors below ±0.000m and four floors above ±0.000m. The height of the superstructure of zone II is ~20.45m, while part of it is 11.25m.

The typical plan view of the BNX is shown in F-16.6-10. The typical cross section of the BNX is shown in F-16.6-11.

The major functional areas housed by the BNX are listed below:

1) Coolant storage and treatment system area;

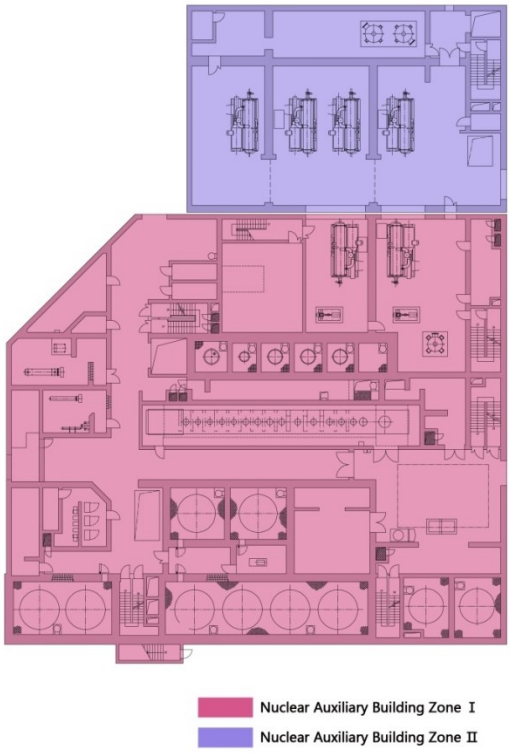
2) Gaseous waste treatment system area.

The BNX is a reinforced concrete structure consisting of walls and slabs. The thicknesses of walls range from 0.3m to 1.1m. The thicknesses of slabs and roofs range from 0.3m to 1.65m.

The horizontal loading is transferred by the roofs and slabs to the walls. The

vertical loading is transferred by walls to the raft foundation. The reinforced concrete walls on the raft provide stiffness for the raft and transfer final loading to the raft foundation.

Detailed safety requirements and the layout of the BNX are provided in *Basis of Safety Case for BNX*, Reference [8].



F-16.6-10 Typical Plan View of the BNX

F-16.6-11 Typical Cross Section of the BNX

d) Diesel Generator Buildings

The diesel generator buildings are comprised of the BDA/BDC/BDU and BDB/BDV. The BDA, BDC and BDU are located on the same raft foundation, with the superstructure forming an integral structure. The same applies for the BDB and BDV. The BDA/BDC/BDU and BDB/BDV are geographically separated by a specified distance to ensure redundancy.

The main geometry of the BDA/BDC/BDU and BDB/BDV is as follows:

1) The length and width of the BDA/BDC/BDU are 44.75m and 26.0m respectively, including five floors above ±0.000m and two floors below ±0.000m;

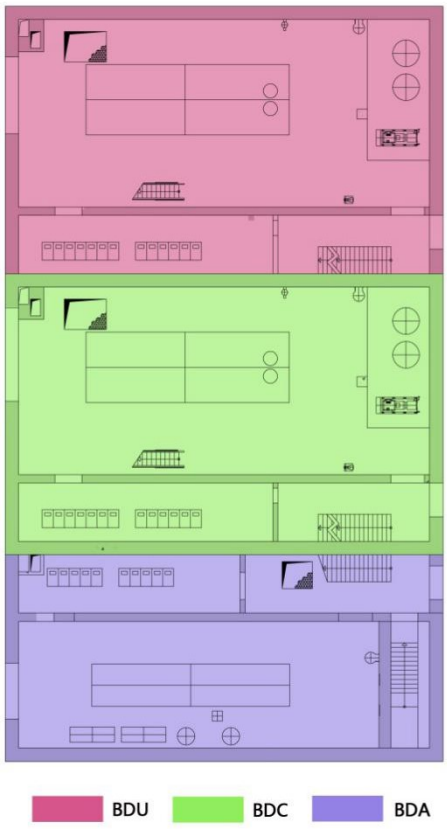
2) The length and width of the BDB/BDV are 28.9m and 26.0m respectively, including five floors above ±0.000m and two floors below ±0.000m.

The typical plan view of diesel generator buildings is shown in F-16.6-12. The typical cross section of diesel generator buildings is shown in F-16.6-13.

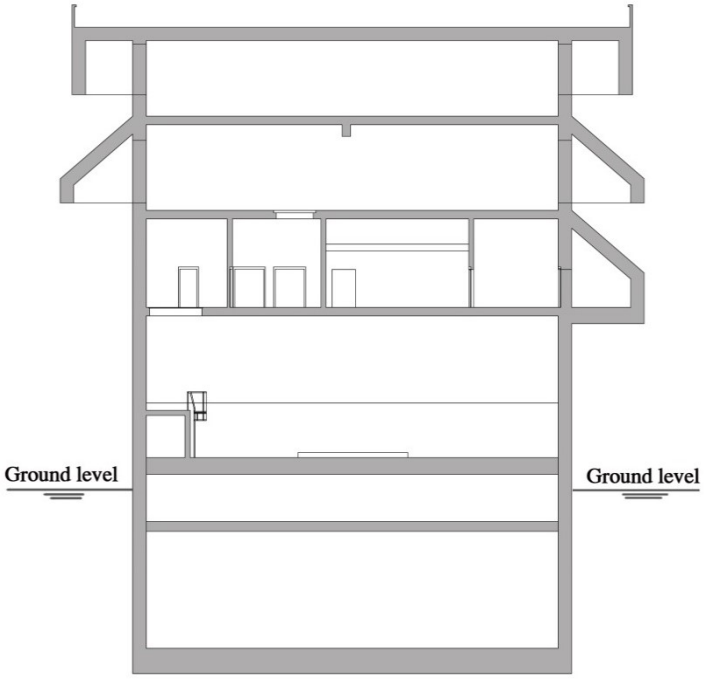
The BDA/BDC/BDU and BDB/BDV are reinforced concrete structures consisting of walls and slabs. The thicknesses of walls range from 0.3m to 0.8m. The thickness of the slab supporting the diesel generator is 1.0m.

The horizontal loading is transferred by the roofs and slabs to the walls. The vertical loading is transferred by walls to the raft foundation. The reinforced concrete walls on the raft provide stiffness for the raft and transfer the final loading to the raft foundation.

Detailed safety requirements and the layout of the diesel generator buildings are provided in *Basis of Safety Case for Diesel Generator Buildings*, Reference [10].



F-16.6-12 Typical Plan View of Diesel Generator Buildings



F-16.6-13 Typical Cross Section of Diesel Generator Buildings

e) BWX

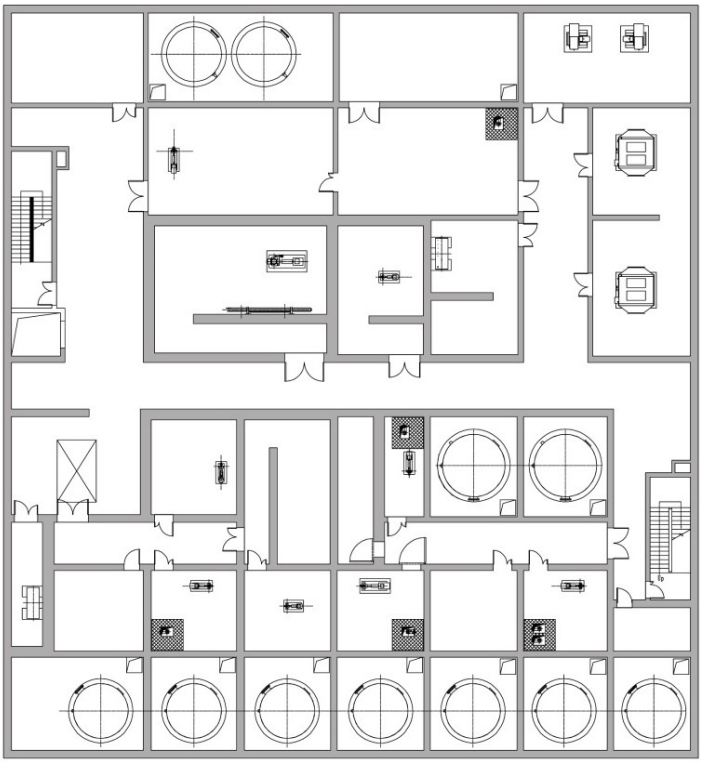
The length and width of BWX are ~48.5m and 44.7m respectively. It has four main levels with two floors above ±0.000m and two floors below ±0.000m.

The typical plan view of the BWX is shown in F-16.6-14. The typical cross section of the BWX is shown in F-16.6-15.

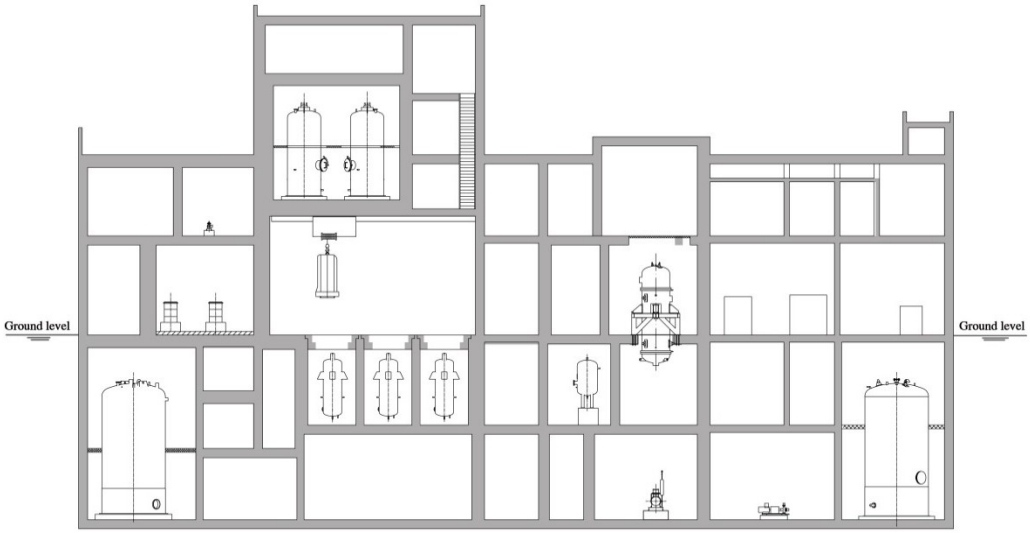
The BWX is a reinforced concrete structure consisting of walls and slabs. The thicknesses of walls range from 0.3m to 0.9m. The thicknesses of slabs and roofs range from 0.4m to 0.9m.

The horizontal loading is transferred by the roofs and slabs to the walls. The vertical loading is transferred by walls to the raft foundation. The reinforced concrete walls on the raft provide stiffness for the raft and transfer the final loading to the raft foundation.

Detailed safety requirements and the layout of the BWX are provided in *Basis of Safety Case for BWX*, Reference [9].



F-16.6-14 Typical Plan View of the BWX



F-16.6-15 Typical Cross Section of the BWX

f) BEJ

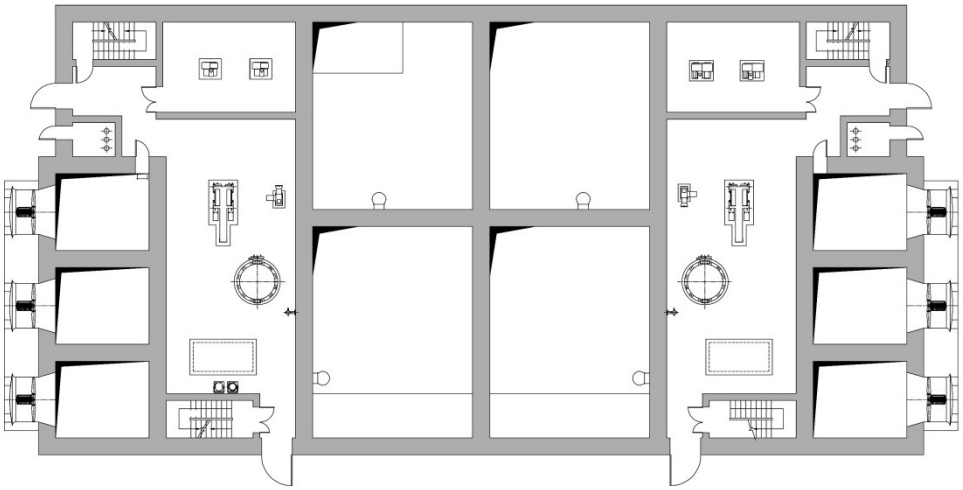
The length and width of the BEJ are ~50m and 26.4m respectively. It has three main levels with two floors above ±0.000m and one floor below ±0.000m.

The typical plan view of the BEJ is shown in F-16.6-16. The typical cross section of the BEJ is shown in F-16.6-17.

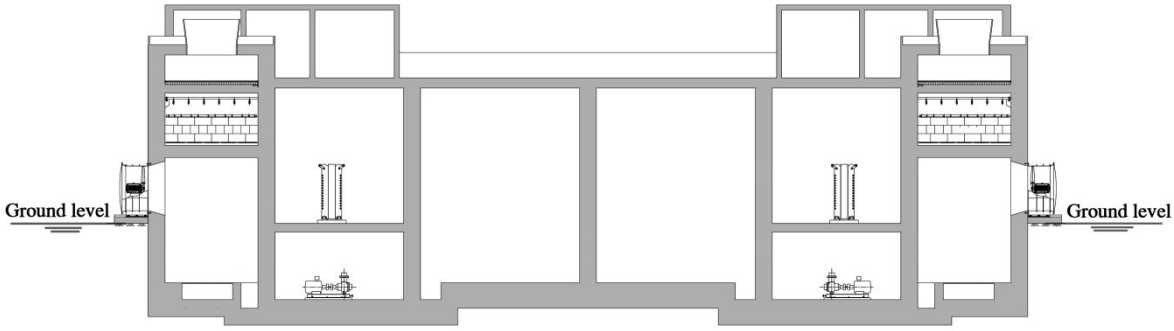
The BEJ is a reinforced concrete structure consisting of walls and slabs. The thickness of outer walls and pool walls is 1.0m. The thickness of slabs at level ±0.000m is 0.5m. The thickness of the roof is 0.5m.

The horizontal loading is transferred by the roofs and slabs to walls. The vertical loading is transferred by the walls to the raft foundation. The reinforced concrete walls on the common raft provide stiffness for the raft and transfer the final loading to the raft foundation.

Detailed safety requirements and the layout of the BEJ are provided in *Basis of Safety Case for BEJ*, Reference [11].



F-16.6-16 Typical Plan View of the BEJ



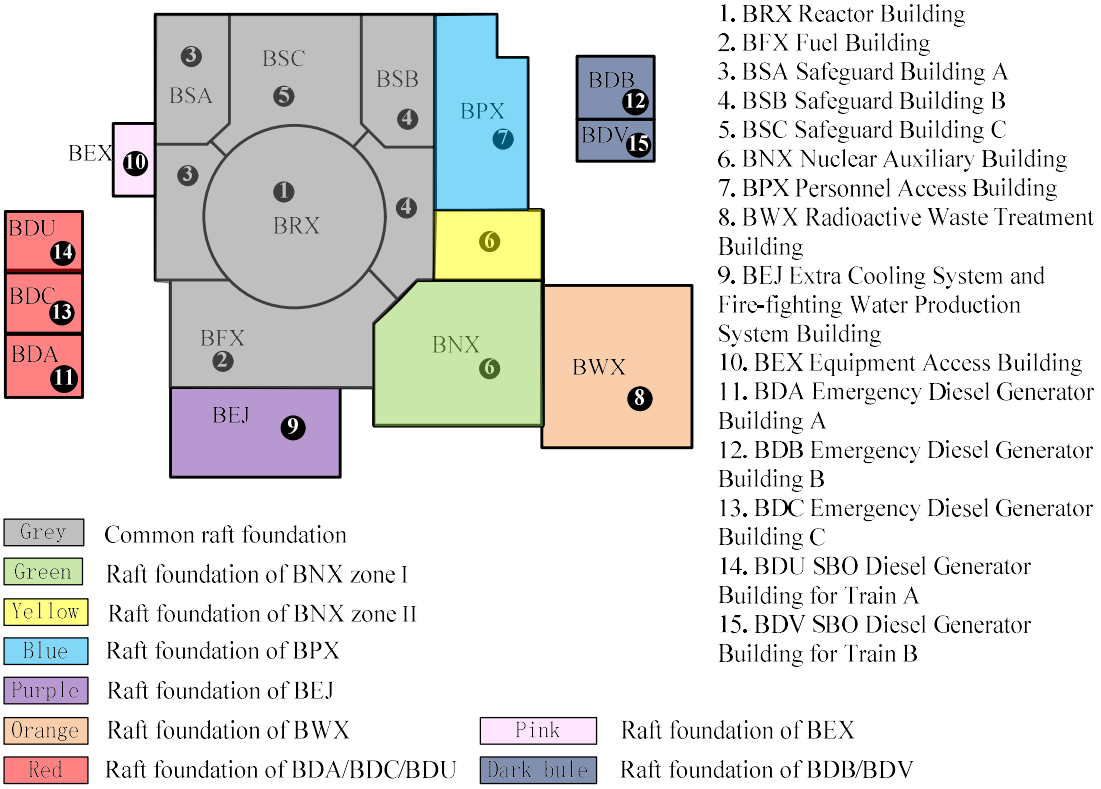
F-16.6-17 Typical Cross Section of the BEJ

16.6.1.4 Foundation

The BRX, BFX and BSA/BSB/BSC are located on a common raft foundation. The BNX, BWX, BEJ, BPX, BEX, BDA/BDC/BDU and BDB/BDV are separately located on independent raft foundations.

Considering that the foundations contact the soil or rocks directly, a waterproof layer is provided surrounding the external surface of the outer walls and the bottom of the foundations.

The raft foundations of the buildings are shown in Figure F-16.6-18.



F-16.6-18 Raft Foundations

a) Common Raft Foundation

The geometry of the common raft foundation is as follows:

1) The length and width of the raft foundation are ~110m and 82m respectively;

2) The thickness of the raft foundation under the BRX is ~4.8m;

3) The thickness of the raft foundation under the BFX and BSA/BSB/BSC is ~3.0m.

The bottom level of the common raft foundation is located at an elevation of -12.600m. There are some pits located in the common raft foundation.

b) Independent Raft Foundations

The BNX is located on an independent raft foundation of which the main geometry is as follows:

1) The length and width of the zone I foundation are ~43.2m and 50.2m respectively;

2) The thickness of the zone I foundation ranges from 1.8m to 2.5m, and part of it is from 1.7m to 2.45m;

3) The length and width of the zone II foundation are 20.8m and 32.1m

respectively;

4) The thickness of the zone II foundation is 2.0m.

For the BNX, the bottom level of the zone I foundation is located at an elevation of -12.750m. The bottom level of the foundation of zone II is located at an elevation of -9.250m.

The diesel generator buildings are separately located on two independent raft foundations. The length and width of the BDA/BDC/BDU foundation are ~44.75m and 31.0m respectively. The length and width of the BDB/BDV foundation are ~31.9m and 33m respectively. The thickness of the foundation is ~1.5m and the bottom level of the foundation is located at an elevation of -12.800m.

The BWX is located on an independent raft foundation. The length and width of the BWX foundation are ~48.5m and 44.7m respectively. The thickness of the foundation is ~1.8m. The bottom level of the foundation is located at an elevation of -12.050m.

The BEJ is located on an independent raft foundation. The length and width of the BEJ foundation are ~26m and 52m respectively. The thickness of the foundation is ~1.5m. The bottom level of the foundation is located at an elevation of -4.000m.

16.6.2 Design Principles

In order to achieve the engineering requirements presented in Sub-chapter 16.3.3, the following design principles are followed in the design process.

The plant is planned to be in operation for 60 years, therefore has a designated “plant design life” of 60 years. From a civil engineering perspective, in addition to the 60 years required for operations, the periods expected for construction and decommissioning also need to be considered in the civil engineering design. To achieve this, the following has been adopted:

a) The civil structures are designed by taking into account construction, the 60 year “plant design life” of operation, and decommissioning. The civil engineering material selected and concrete cover design would have a design life of 100 years and the external hazards for normal operations are based on 100 year return period.

b) 65 years has been adopted for relaxation of the inner containment tendons, taking into account an additional 5 years of construction period only, as the fuel will be definitively removed from the vessel during the decommissioning phase.

The principal construction materials for civil structures are concrete, reinforcing steel, prestressing tendons and structural steelwork. The materials selected are widely

available in the UK and conform to the applicable design requirements and associated specifications.

Based on the design codes and standards listed in Sub-chapter 16.4, a conservative method and complete load list are adopted to analyse and design in order to ensure that a reliable result achieved can satisfy the design requirements. The durability design is also taken into consideration for normal operation.

The software used in the analysis and design for the civil structures in the UK HPR1000 are listed below:

a) ACS SASSI: commercial Windows software which is used to undertake linear and non-linear three-dimensional seismic SSI analysis.

b) ANSYS: commercial finite element software which is used to undertake modelling and analysis of the civil structures.

c) ABAQUS: commercial finite element software which is used to undertake the analysis of beyond design basis loading of the containment.

d) LS-DYNA: commercial software which is used to undertake the non-linear analysis of malicious aircraft impact, dropped loads impact etc.

e) REINCAL: custom software, internally-developed by the Requesting Party (RP), which determines the section and reinforcement of reinforced concrete structural members.

The verification and validation of the commercial software refers to section 9 of report *Structural Analysis and Design Method Statement*, Reference [35]. The verification and validation of the software are conducted to ensure that the analytical modelling and design calculation results give an accurate representation of the UK HPR1000 structural behaviour.

16.6.3 Loads and Load Combinations

The loads used in the design of the civil structures are derived from normal operation, internal hazards, external hazards and test loads. Normal operational loads conservatively reflect the maximum loads encountered during normal plant operation and shutdown. The external hazards are derived in Chapter 18 Sub-chapter 18.6 and the internal hazards are derived in Chapter 19 Sub-chapter 19.6.

16.6.3.1 Normal Operational Loads

Normal operational loads include dead loads (D), live loads (L), liquid loads (F), earth pressures (H), prestressing loads (F_p), crane loads (C_{cr}), normal operating temperature loads (T_o), normal operating equipment reactions (R_o) and external pressure load (P_v).

a) Dead Loads (D)

Dead loads include the weight of the structure and the weight of the fixed piping and equipment during the normal operation of the plant.

b) Live Loads (L)

Live loads are divided into live loads during construction and during normal operation. Live loads during construction (carried loads, support imposed loads, etc.) are specified according to the construction conditions. Live loads during normal operation are produced by the use and occupancy of the buildings.

c) Liquid Loads (F)

For the water pools within the Nuclear Island (NI), hydrostatic loads caused by a depth of water in the pools are treated as liquid loads (F).

d) Earth Pressures (H)

Earth pressures are derived from the effect of backfilled soil and exert loads on side walls which are not protected by another building.

e) Prestressing Loads (F_p)

Prestressing loads are the loads resulting from the application of prestressing which is used in the design of the prestressed concrete internal containment.

f) Crane Loads (C_{cr})

The dead load components of the crane loads are incorporated within the dead load (D) estimation. The crane load-rated capacity, C_{cr}, is applied as a separate load case in detailed local models in accordance with the variability and probability of occurrence.

g) Normal Operating Temperature Loads (T_o)

Normal operating thermal loads are thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

h) Normal Operating Equipment Reactions (R_o)

Normal operating equipment reactions are piping and equipment reactions, or related internal moments and forces that occur under normal operating and shutdown conditions, excluding dead loads and earthquake reactions.

i) External Pressure Loads (P_v)

The annulus between the internal and external containments is designed to be at a negative pressure relative to the ambient pressure. The pressure within the internal containment during normal operation and negative pressure of annulus generate the external pressure loads (P_v).

j) Ground Water Level

There are two assumptions for ground water level. One is at ground level and the other is at the bottom level of the common raft foundation.

16.6.3.2 Loads from Internal Hazards

Loads arising from internal hazards include internal flooding, postulated high energy pipe failures, dropped loads, internal missiles, internal explosion and vehicular impact.

a) Internal Flooding (F_a)

Internal flooding can occur due to failure of pipework or tanks containing liquid. For compartments within the NI, the potential internal flooding is treated as a static load.

b) Postulated High Energy Pipe Failures

The effects caused by the failures of high energy pipes include overpressure, high temperature, pipe whip, jet impingement, flooding and blast.

1) Accident Pressure Load (P_a)

The design of the civil structures considers a differential pressure across the compartment or building components (such as walls or slabs).

2) Accident Temperature Load (T_a)

The accident temperature load is caused by temperature distributions within the concrete structure occurring as a result of accident conditions generated by a postulated pipe break and includes T_o.

3) Accident Pipe Reactions (R_a)

The accident pipe reactions are considered under thermal conditions generated by a postulated pipe break and includes R_o.

4) Jet Impingement (Y_j)

The accident jet impingement load is generated by the postulated pipe breaks, and has an effect that is limited to a small area.

5) Missile Impact (Y_m)

The accident missile impact load is generated by the postulated pipe breaks such as pipe whip and the effect is limited to a small area.

6) Reaction of the Broken Pipe (Y_r)

The accident reaction force is generated by the postulated pipe breaks.

7) Blast (Y_r)

The accident blast wave is generated by the postulated pipe breaks and has an effect that is limited to a small area.

c) Dropped Loads (A_m)

A dropped load occurs if the lifting device can no longer control the load on the hook. Moreover, the lifting devices have the potential to fall or collapse according to the standards applicable to the lifting devices.

d) Internal Missiles (A_m)

Internal missiles may be generated by high energy component failures and rotating equipment.

e) Internal Explosion (A_{exp})

The main potential sources of internal explosion include explosive gases released from a regular operational process, explosive gases released as a result of a pipe failure and high energy arcing faults.

f) Vehicular Impact (A_m)

Vehicular impact may be considered as a type of missile impact.

16.6.3.3 Loads from External Hazards

Loads from external hazards include earthquake, aircraft impact, wind and tornado, temperature, snow and icing, and external explosion.

a) Earthquake (E_{ss})

E_{ss} is defined as a Safe Shutdown Earthquake (SSE) which certain SSCs in the plant are designed against to ensure that they remain functional.

b) Aircraft Impact

Aircraft impact considered in GDA is divided into two types, accidental aircraft impact and malicious aircraft impact. Accidental aircraft impact is considered as a design basis event. Malicious aircraft impact is considered as a beyond design basis condition.

c) Wind (W)

Wind loading is determined using the wind speed value (3-second gust).

d) Tornado Load (W_t)

Tornado loads, including the effects of missile impact, are a combination of the following:

1) The tornado generated missile impact effects (W_{tm}).

2) The differential pressure loads due to rapid atmospheric pressure change (W_{tp}).

3) The loads due to tornado wind pressure (W_{tq}).

e) Temperature

Extreme ambient temperatures include extreme high or low temperatures. These temperature values are used to calculate the thermal effect on the civil structures based on the temperature gradient.

f) Snow (S) and Icing (D_i)

Snow and icing loads are considered to be live loads.

g) External Explosion

An external explosion may take the form of a deflagration, which generates moderate pressures, heat or fire, or a detonation which generates high pressures and thermal effects.

h) External Flooding

The external flooding is highly site specific. The probable influence of flooding for civil structures will be considered in the site specific stage.

16.6.3.4 Test Loads

Test loads are loads which are applied during structural integrity or leak rate tests of prestressed concrete containment, which include:

a) Pressure during the structural integrity and leak rate tests (P_t);

b) Thermal effects and loads during the test (T_t).

16.6.3.5 Load Combinations

To achieve the design requirements introduced in Sub-chapter 16.3, the civil structures are analysed and designed under the loads arising from normal operations, internal hazards, external hazards and test loads. The individual loads applied to the nuclear safety-related buildings are listed in Table T-16.6-1.

The individual load is multiplied by a corresponding load factor. These factored loads are combined to make different load combinations. The load combinations considered in the design of concrete structures comply with Reference [16]. The load combinations considered in the design of the prestressed concrete containment comply with Reference [14]. The detailed load combinations for the civil structures are specified in method statement reports, References [35], [36], and [37].

T-16.6-1 Loads Applied to Nuclear Safety-Related Buildings

Building		BRX Internal Structures	Internal Containment	External Containment	BFX	BSA/BSB/ BSC	BNX	BWX	BDA/BDB/BDC /BDU/BDV	BEJ
Normal operational loads	Dead loads (D)	×	×	×	×	×	×	×	×	×
	Live loads (L)	×	×	×	×	×	×	×	×	×
	Liquid loads (F)	×		×	×	×				×
	Static earth pressure (H)				×	×	×	×	×	×
	Prestress loads (F_p)		×							
	Normal operating thermal loads (T_o)	×	×	×	×	×	×	×	×	×
	Normal operating equipment reactions (R_o)	×	×	×	×	×	×	×	×	×
	External pressure loads (P_v)		×							
Loads from internal hazards	Internal flooding (F_a)	×	×	×	×	×	×	×	×	×
	High energy pipe failure	×	×	×	×	×	×	×		
	Vehicular impact(A_m)				×	×				
	Dropped loads (A_m)	×			×	×	×	×		
	Internal missiles (A_m)	×			×	×	×	×	×	
	Internal explosion (A_{exp})	×			×	×	×		×	
Loads from external hazards	Snow and icing (S)			×	×	×	×	×	×	×
	Wind (W)/Rain (R)			×	×	×	×	×	×	×
	Tornado load (W_t)			×	×	×	×	×	×	×
	Earthquake (E_{ss})	×	×	×	×	×	×	×	×	×

Building		BRX Internal Structures	Internal Containment	External Containment	BFX	BSA/BSB/ BSC	BNX	BWX	BDA/BDB/BDC /BDU/BDV	BEJ
	Temperature	×	×	×	×	×	×	×	×	×
	External explosion			×	×	×	×	×	×	×
	{ }			{ }	{ }	{ }	{ }	{ }	{ }	{ }
Test loads	Pressure (P_t)		×							
	Thermal effects (T_t)		×							

Note: The symbol “×” means that the load is included as a design input for the structure.

16.6.4 Methods of Analysis and Design

From the perspective of structural characteristics of buildings, the buildings include three types, prestressed containment, normal reinforced concrete structures and raft foundations. The methods of analysis and design of these three types of buildings are presented as follows.

16.6.4.1 Prestressed Containment

The design of the internal containment is required to satisfy the design basis condition as well as the design extended condition. The design basis condition covers normal operation, design basis internal and external hazards. This is achieved based on a codified design approach. The design extended condition covers events more severe than those considered in the design basis condition, in order to demonstrate that there is not a disproportionate increase in risk. The design extended condition is generally based on beyond design basis approaches.

The design procedure for the design basis condition includes global and local structural analysis and durability design. The global structural analysis of the prestressed containment generates the overall response of the structure and provides the distribution of the internal forces, moment and deflection of structural members. The results of the global analysis are evaluated to confirm member section sizes and quantities of reinforcement based upon the specified codes and standards. For local areas of the prestressed containment with penetrations that are not included in the global containment model or local areas which are coarsely modelled in the global containment model, refined results are obtained from more detailed analyses.

During the analysis and design, the durability design requirements in terms of exposure categories and classes are evaluated. Based on the model of the containment used for global structural analysis, the tendon tensioning sequence evaluation and tendon failure evaluation are carried out.

In addition to the design basis condition, it is required for nuclear structures that there is not a disproportionate increase in risk from events more severe than the design basis event. This is addressed by consideration of the design extended condition. For the internal containment structure, it is required to understand the margin by predicting the internal pressure capacity above the internal pressure for the design-basis Loss of Coolant Accident (LOCA), with the higher pressure and temperature loads that may occur under severe accident conditions from various severe accident scenarios.

This is achieved by deterministic analysis of the ultimate capacity of the internal containment structure based on the selected severe accident scenarios set out by the Severe Accident discipline. This information is to be used as the input data for the overall Probabilistic Safety Analysis (PSA) of the containment fragility assessment.

More information about the methods used for the analysis and design of the prestressed containment is documented in the report *Internal Containment Analysis and Design Method Statement*, Reference [37].

16.6.4.2 Normal Reinforced Concrete Structure

The methodology used for the analysis and design of normal reinforced concrete structures is based upon the ACI 349 nuclear-specific code, Reference [16]. The analysis and design includes global and local structural analysis, serviceability and durability design.

The global structural analysis aims to calculate the overall response of the structure and provides the internal forces, moments and deflections of structural members. The global structural analysis and design involves finite element modelling of the structural members, load application to model, load combinations and calculation, finite element analysis and structural member design. The results of the global structural analysis are evaluated to confirm member section sizes and quantities of reinforcement based upon the specified codes and standards.

Local structural analysis is used for non-standard zones which are not applicable to the REINCAL software and local areas with new local loads which are not considered in global structural analysis. The local structural analysis is undertaken in additional calculations by hand or using software, or by development of the global analysis model with a more refined finite element mesh size. The latter involves the transfer of results from the global structural analysis to a local structural analysis.

The serviceability and durability design are incorporated in the analysis and design process. The serviceability and durability design include the determination of exposure categories and classes, thickness of concrete cover, deflection checks, crack width calculations etc.

Detailing of reinforcement and concrete is consistent with the design assumptions and with best practices for seismic resisting construction. The good practice for detailing includes careful treatment of intersections, sufficient anchorage and overlap length, careful treatment of openings, etc.

More information about the methods used for the analysis and design of normal reinforced concrete structures is documented in the report *Structural Analysis and Design Method Statement*, Reference [35].

16.6.4.3 Raft Foundation

The raft foundation is the structural element that serves as the interface between the building superstructures and the soil. The raft foundation needs to support the superstructure and ensure the stability of the buildings. The raft foundation also serves as structural barrier against radioactive release to the ground and water ingress. Measures are taken to meet the requirements of serviceability, such as crack limit and

waterproofing.

The generic subgrade conditions in GDA are Soft and Medium ground conditions. For the GDA study, the main site information available to characterize the ground is the assumed shear wave velocity (V_s). A series of credible values of geotechnical parameters are defined correlated to the declared shear wave velocity range in the *Generic Site Report,* Reference [38], including the stiffness and the allowable bearing capacity.

The soil/structure systems are modelled using soil springs. The model is used to ensure that the allowable bearing capacity is not exceeded, and to analyse the foundation response in terms of differential settlement and inclination. The generic vertical and horizontal ground spring values for each building are calculated by iteration analysis based on shear wave velocity ranges.

The global finite element model which includes the raft and superstructures is used to design the raft foundation. The raft foundation analysis and design adopts the results from the global structural analysis and seismic analysis.

The overall stability analysis evaluates the stability against sliding, overturning, and flotation. The global stability design has been undertaken in accordance with the codes ACI 349M-13, Reference [16] and NUREG-0800, Reference [26]. The stability of the raft foundation ensures that all structures located on top of it are stable.

Sliding stability is checked by using equivalent static evaluation which compares the equivalent static sliding force with static resistance. This evaluation approach is conservative as it is assumed the sliding force is applied continuously. Similar to sliding check, the analysis of overturning stability also adopt a conservative equivalent static approach. The analysis of flotation stability considers the potential for flotation of the raft and structure under the buoyancy by the ground water or flood. The factor of safety against flotation equals to the resisting force divided by buoyant force.

The width of seismic joints between neighbouring buildings is checked according to the method provided in Section 7.3 of ASCE 43-05, Reference [23]. The horizontal deformation under seismic loading is combined with the horizontal deformation caused by inclination of the foundation and superstructures during the seismic event. The seismic joint assessment uses the absolute sum of these two values to calculate the horizontal deformation of each building. The calculation results show that the width of the seismic joint is greater than the corresponding minimum separation of buildings for each seismic joint, which means that the neighbouring buildings have no adverse interaction.

More information about the methods used for the analysis and design of the raft foundations is documented in the *Raft Foundation Analysis and Design Method Statement*, Reference [36].

16.6.5 Substantiation

The nuclear safety-related structures are substantiated to be robust and achieve the engineering requirements which satisfy the Safety Function Requirements (SFRs).

a) Global stability

The overall stability of the raft foundation has considered resistance against sliding, overturning and flotation stability, incorporating appropriate factors of safety and ensuring that resisting forces and moments are more than, or at least equal to, the design actions.

b) Strength design

The strength design ensures the structures and structural members are reliable and robust, and that structures and structural members are designed to have the required strength during the normal and fault conditions while remaining within their elastic stage.

The design strength and required strength of the members have been calculated according to the load combinations. The structures and structural members have been designed to have strength at least equal to the required strength in flexure, axial force, shear and torsion.

c) Concrete barrier and non-barrier evaluation

The global effects and local effects caused by internal hazards are evaluated for the reinforced concrete barriers and non-barriers. Global effects are analysed either with simplified methods or using finite element methods. Local effects on concrete structural members include penetration, scabbing, perforation and through thickness cracking.

For the internal hazard cases, based on the analysis results of global and local effect, the barriers and non-barriers have met the engineering requirements.

d) Serviceability design

The serviceability design of the buildings and corresponding structural members has considered stress limitation, crack control and deflection control.

The results of the crack width assessment for each structural member confirm that crack widths for all reinforced concrete structural members are limited and comply with the requirements of the applicable codes and standards. Reinforced concrete slabs and beams subjected to flexure have been designed to have adequate stiffness to limit deflections and deformations that adversely affect the strength or serviceability of structural and non-structural members. The minimum design thickness of all concrete slabs and beams meets the minimum thickness requirements specified in the applicable codes and standards.

e) Air / Water tightness

The air tightness for the internal containment is provided by two barriers. The first barrier is a steel liner and the second barrier is a prestressed concrete structure whose airtightness is achieved by controlling the concrete stress to limit the development of cracks and by adding structural inspection measures.

The water tightness of pools is achieved by two barriers. The first barrier is an internal stainless steel liner. The steel liners of important pools are equipped with a leakage detection system. The concrete structural members are used as the second barrier by controlling concrete cracks. The calculations of crack width conform to the requirements in BS EN 1992-1-1 section 7.3.1.

f) Durability

The durability design considered for GDA includes the requirement of concrete strength, concrete cover to the reinforcement, and crack width limitation of concrete for different classes and grades.

Cover to reinforcement for corrosion and fire protection has been taken as the most onerous value of the specified cover determined from applicable codes and standards.

16.7 Beyond Design Basis Conditions

Beyond design basis conditions are identified according to the fault sequences that have lower frequencies than that of the design basis. The beyond design basis evaluation includes the ultimate capacity evaluation for internal containment, malicious aircraft impact evaluation and cliff-edge effect evaluation of extreme environmental hazards.

16.7.1 Ultimate Capacity Evaluation for Internal Containment

16.7.1.1 Purpose

The purpose of the ultimate capacity evaluation for the internal containment is to prove the conservatism and reliability of the internal containment under the beyond design basis pressure.

16.7.1.2 Evaluation Method

This evaluation is carried out through a non-linear analysis methodology using a three-dimensional finite element simulation, which includes modelling, establishing the constitutive relations of materials, loading, global analysis of structural reactions, analysis of typical locations of containment, determination of failure modes and criteria, and evaluation of the ultimate capacity of containment according to the failure criteria.

a) Modelling

ABAQUS finite element analysis software is used to establish an overall model of the containment. The model includes the prestressing, steel liner, rebar, concrete, typical members and discontinuity parts (such as the equipment hatch and personnel hatches, ribs, etc.). Appropriate elements are used to simulate different materials. Concrete is modelled using solid elements, the steel liner is modelled as a skin covering the surface of the concrete solid elements, the reinforcement is modelled using surface elements, and the prestressing is modelled using beam elements.

b) Establishing the Materials Constitutive Relations

Establishing the constitutive relations of materials by considering their non-linear characteristics is carried out according to UK materials, which is then used as the input data in the model.

c) Loadings

The permanent loading and prestressing loading are applied in the first analysis step, and then a step-by-step analysis applying the pressure loading is carried out in the second analysis step. In the last step, the thermal loading is applied.

d) Analysis

In accordance with the calculation results, a global analysis of structural reactions and an analysis of typical locations of containment are carried out to evaluate the structural behaviour and performance.

e) Criteria and Evaluation

Based on Reference [39], failure modes and criteria are determined. The ultimate capacity of the containment can be evaluated according to the calculation results and failure criteria.

Except for the above finite element ultimate capacity analysis, additional calculations by hand are undertaken to supplement and validate the finite element (FE) analysis. The additional calculations by hand include:

a) Shear checking for the equipment hatch;

b) Concrete checking around the stiffener of the equipment sleeve;

c) Shear checking for the gusset;

d) Main failure mode checking for the standard zone.

16.7.1.3 Conclusion

Through FE ultimate capacity analysis and hand calculations, the ultimate bearing capacity of the internal containment is evaluated. The safety margin of the internal containment is calculated and the conservatism and reliability of the internal

containment design are proved.

16.7.2 Aircraft Impact Evaluation

16.7.2.1 Safety Function

The following functional requirements are incorporated in the plant design:

a) The reactor core remains cooled, or the containment remains intact.

b) Spent fuel cooling or spent fuel pool integrity is maintained.

16.7.2.2 Scope of Protection

The civil structures which are physically protected against aircraft impact are shown in the report *Basis of Safety Case for Aircraft Impact*, *Reference [40]*.

16.7.2.3 Evaluation Method

For malicious aircraft impact, the global effects, local effects, vibration effects and fire effects of the aircraft impact are considered. {

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16.7.2.4 Conclusion

The aircraft impact assessment of the UK HPR1000 has been undertaken in accordance with internationally recognised and accepted nuclear-specific codes and standards, providing a civil engineering design that meet the intended reliability and functional requirements.

From the civil engineering perspective, the calculation and analysis results show that the containment remains intact, the integrity of spent fuel pool is maintained, and the integrity of structures supporting the systems and components of reactor core cooling and spent fuel cooling is maintained.

16.7.3 Cliff-Edge Effects for Extreme Environmental Hazards

16.7.3.1 Purpose

The purpose of the analysis of cliff-edge effects is to provide demonstrable evidence that the civil engineering structures subjected to design basis environmental hazards do not have any “cliff-edge” effects, so that a small change in the definition of the design basis hazards will not lead to a disproportionate increase in radiological consequence. This has been achieved by demonstrating that there are sufficient remaining margins in the civil design under extreme environmental hazards.

16.7.3.2 Evaluation Method

The extreme environmental hazards for the civil engineering structures are identified and screened. The earthquake, low and high air temperature, extreme snow, external explosion are considered for the “cliff-edge” evaluation.

The cliff-edge evaluations for the civil engineering structures are undertaken on a hazard-by-hazard basis. The seismic hazard is expected to be the primary and likely to be bounding natural hazard for the civil design. Hence, a detailed cliff-edge evaluation is provided for the seismic hazard, and for other hazards other than seismic, a small increase of snow loading, air temperature and explosion beyond design basis will not lead to a sudden collapse of the building, nor significant increase of pounding risk, and therefore will not lead to a disproportionate increase in radiological consequence.

For earthquake, the UK HPR1000 select 150% DBE as seismic target to withstand, and this is evaluated with the lower limit of 10% confidence of median ground acceleration. A seismic margin assessment is based on the EPRI’s seismic margin methodology described in *A Methodology for Assessment of Nuclear Power Plant Seismic Margin* (NP-6041-SL). In the assessment, a significant design margin is demonstrated for bearing capacity. The hierarchy of failure provides a discussion on a probable failure mode of the structure when subject to a much larger earthquake.

For internal hazards, the cliff edge effect has already been considered in the derivations of bounding cases of internal hazards.

16.7.3.3 Conclusion

Based on the analysis described above, it is demonstrated that the civil engineering structures do not have “cliff-edge” effects for design basis environmental hazards, so that a small change in the definition of design basis hazards will not lead to a disproportionate increase in radiological consequence.

16.8 Construction and Testing

Aspects that are fundamental to underpin the safety case for construction and testing of the civil structures are taken into consideration. CDM regulations are also considered appropriately for the design and construction.

16.8.1 Constructability

The nuclear safety-related buildings are conventional slab-wall-structures. The construction of the reactor building containment structures, common raft foundation and BFX roof are representative to show the constructability of the buildings.

a) Containment

The reactor building containment structures are circular in shape and have cylindrical walls with a dome above the common raft foundation. The internal containment is a prestressed concrete structure and the external containment is a reinforced concrete structure located outside of the internal containment.

The containment structures are constructed layer by layer. The construction of internal containment precedes the external containment by a height difference to provide sufficient space for the formwork construction. The steel liner of the internal containment structure is used as the internal formwork for the concrete pouring.

Some construction steps need to be performed in a certain order to improve constructability, such as installation of polar crane corbels on the upper part of the internal containment and hoisting of the dome steel liner.

After the concrete construction of the internal containment is completed, tensioning of the pre-stressing tendons needs to be carried out. The force applied to the pre-stressing tendons and the corresponding elongation during the tensioning operation is measured with indicators and recorded.

b) Common raft foundation

There are significant quantities of steel reinforcement bars and embedded pipes/plates in the common raft foundation. A 3D simulation model is used to avoid the potential clash between the steel reinforcement and embedded items.

The construction difficulties for the common raft foundation include concrete temperature control and crack prevention, and the quality control of the concrete pouring and curing process. The internal temperature of concrete is monitored to guide the concrete pouring and curing process in time and effectively.

c) BFX roof

The upper part of the fuel handling hall is a large space structural system, which cannot provide full support for the roof slab construction. Therefore, the structural design of the roof slab consists of steel beams, profiled steel plates and a cast-in-situ concrete roof slab.

The roof slab of BFX is 1000mm thick and is a layered reinforced concrete slab. For the first stage, a 300mm thickness of concrete is poured, and profiled steel plates and steel beams act as formwork to support the concrete while it cures. When the 300mm thick concrete section reaches the design strength, a composite beam is formed with the profiled steel plates and steel beams, and then the upper 700mm thickness of concrete is poured to form the 1000mm thick reinforced concrete roof slab.

16.8.2 Testing during Construction

a) Pre-stressing test

Pre-stressing testing includes a pre-stressing tendon friction test and a pre-stressing tendon full-scale grouting test.

Before the pre-stressing tendons are tensioned for construction, a friction test of the pre-stressing tendons is carried out to calculate the actual friction coefficient between the tendons and the ducts based on measured elongation values. The pre-stressing tendon full-scale grouting test is carried out before the final grouting to ensure that the grouting process meets the design requirements.

b) Structural integrity test of containment

The structural integrity test of the containment, also known as the strength test, is performed to demonstrate the construction quality and ensure the structural integrity of the containment structures in the case of a LOCA.

The test is conducted before the start of operation of the NPP. Structural monitoring data is collected by the containment instrument system during the construction period. The strains, displacements, temperatures, and pressures in the containment structures are recorded.

The acceptance criteria for the strength test comply with the requirements of CC-6000 of ASME BPVC, Section III, Division 2, Reference [14].

c) Leak tightness test for the Spent Fuel Pool

The Spent Fuel Pool (SFP) contains a stainless steel liner. After the liner has been installed, a leak tightness test is carried out by filling the SFP with water for 15 days. The SFP is considered to be leak tight if no leak is detected via the drainage network.

d) Material testing

Concrete material testing includes identification of the raw materials, acceptance and inspection tests for the water, cement, aggregates, admixtures and external additives, and preliminary tests for the mix ratios of concrete, feasibility test and quality inspection experiment. Rebar material testing includes rebar procurement qualification test and site acceptance test.

The conventional health and safety risks to workers and the public that may arise during the construction, commissioning, operation, maintenance, and decommissioning of civil engineering structures 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 Chapter 25. The risk management processes are recorded in Conventional Health and Safety Design Risk Registers for the civil engineering structures.

16.9 In-Service Inspection and Testing

An effective EMIT programme is essential for the safe operation of the NPP. The ageing and degradation effects on the civil engineering structures are monitored and suitable measures are taken to ensure that the levels of reliability and availability of all civil engineering structures important to safety remain in accordance with the design requirements over the lifetime of the NPP.

16.9.1 Building Instrumentation

The instrumentation for the buildings consists of settlement instrumentation for the buildings and containment instrumentation.

a) Settlement instrumentation

Topographic survey control points are set to monitor the absolute settlement of the buildings and the differential settlement between the neighbouring buildings during the operation of the NPP.

b) Internal containment instrumentation

The sensors of the Containment Instrumentation System (EAU) monitor the global deformation of the internal containment structure, concrete strain, temperature of the concrete and air, and condition of the pre-stressed tendons.

1) Global deformation of the internal containment

The global displacement of the internal containment structure includes horizontal diametrical variations and height variations of the containment cylinder, and curve deformations of the containment dome which can be measured with a plumb line system and an invar line system respectively.

2) Concrete strain

Strain gauges are embedded in the concrete of the internal containment structure and are used to measure the concrete strain during construction, containment testing and operation. Temperature measurements are used to correct measured strain values for thermal effects.

3) Pre-stressed tendon instrumentation

Tension measurements are undertaken on the vertical pre-stressed tendons protected by the flexible fill (paraffin wax or mineral oil). Dynamometers are situated on the passive side of the pre-stressed tendon. During the whole measuring process, the containment instrument system data collection room should record the data continuously.

To monitor potential tendon failures in the internal containment structure, strain gauges are embedded in the concrete in zones around the grouted tendons.

16.9.2 Structural Integrity Test of Containment

During normal operation, the structural integrity test of the containment is conducted periodically during shutdown and refuelling periods.

The maximum pressure applied for the structural integrity tests during the NPP in-service phase is lower than the maximum pressure applied for the structural integrity test before the start of operation of the NPP (i.e. the first structural integrity test of containment). Apart from the maximum applied pressure, the periodic structural integrity test during the NPP in-service phase is consistent with the first structural integrity test.

16.9.3 Containment Leak-tightness Test

The containment leak-tightness test comprises the integrated leakage rate test of the containment system and the local leakage rate test measured individually for containment components (penetrations, isolation valves, equipment hatch, personnel hatch). The test is performed to verify whether the leak rate is lower than the maximum allowable leakage requirement.

The first global leak-tightness test is conducted after the first structural integrity test. During normal operation, the test is conducted periodically during the shutdown and refuelling periods.

16.9.4 Routine Inspect and Maintenance

During the operation of NPP, routine checks, inspections and maintenance for the civil engineering structures are conducted to monitor and mitigate the ageing and degradation effects on the civil engineering structures. The usual/common methods and tools are used. The visual inspection is the basic procedure and could be complemented with other methods or instrumentation when relevant.

The checks and inspections for the civil engineering structures primarily include:

a) Steel structures checks.

b) Check of the internal containment steel liner.

c) Water-proofing check of roof and floor; water drainage checks for roofs.

d) Water tightness check of pools with steel liners.

e) Painting system checks.

f) Crack checks for concrete walls and slabs.

16.10 ALARP

This sub-chapter summarises the systematic review carried out against UK RGP, Operating Experience (OPEX) and takes cognisance of risk insights where relevant. Based on the review results, this sub-chapter lists the gaps or high risks, and provides links to the lower tier supporting documents that address them.

The general ALARP methodology for the UK HPR1000 is described in the report *ALARP Methodology*, Reference [43], and the detailed ALARP demonstration for civil engineering refers to Reference [44].

16.10.1 Holistic ALARP Assessment

16.10.1.1 Evolution of the HPR1000

The HPR1000 technology has a strong pedigree. It has been developed based on the designs, construction, operations and maintenances experience from a series of successful Chinese commercial NPPs.

Comparing to ACPR1000, the main improvements for the HPR1000 civil engineering structures include:

a) Single-unit layout scheme.

b) Double containment.

c) Protection against large commercial airplane crash.

d) Safe shutdown earthquake with a peak ground acceleration of 0.3g.

16.10.1.2 Compliance with RGP

RGP in this context should comprise practices that have been approved for use in nuclear reactor plant designs by recognised authoritative bodies and are considered relevant and appropriate to the UK HPR1000 design.

RGP is typically defined in the following non-exhaustive list of sources of which the detailed RGPs refer to the section 3.2 in Reference [44].

a) Legislations and Regulations

b) Safety Assessment Principles (SAPs) and Technical Assessment Guides (TAGs) produced by the Office for Nuclear Regulation (ONR)

c) International Atomic Energy Agency (IAEA) Safety Standards

d) Recognised design codes and standards (e.g. British Standards, Eurocodes, US codes and standards)

The consistency review work is carried out based on the requirements from RGPs for the overall design of civil works and structures, including the suitability and compliance analysis of codes and standards, and application of US nuclear design codes and standards in the UK.

The review shows that the safety function requirements and engineering requirements of civil works of HPR1000 mainly comply with RGP. {

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16.10.1.3 OPEX Review

Although the OPEX from UK and other countries (e.g. US, Europe and Asia) are not formally RGP, these OPEX are a valuable contribution to demonstrating that the design has taken into account wider learning. In particular, it has the potential to identify actual events that have influenced the design and to identify design options for consideration.

Based on the review methodology for OPEX, Reference [45], the civil engineering technical area has identified applicable OPEX obtained from a list of potential OPEX

sources. OPEX including the definition of concrete exposure class, concrete cover of reinforced concrete members and quality control of concrete materials have been considered in the design of civil engineering structures.

16.10.1.4 Risk Assessment

The insights from risk assessment could inform the design, including any aspects of the design where the contribution to risk is elevated and could therefore be a candidate for improvement via changes to the design. Risks insights which could lead to the identification of further potential improvements have been transferred to the civil engineering discipline.

A typical case of improvements resulting from a risk assessment is the overall modification of BFX. The overall BFX geometry change is synthesised from the solutions. The ALARP review of civil engineering for the BFX geometry change is incorporated in the solutions as one of the aspects to be considered. The effect of the design change on the civil engineering is evaluated. The impact on the global behaviour of the BFX is acceptable and no changes in the overall design are necessary. More information about the BFX modification is documented in the report *Impact Analysis of Design Modification on Civil Engineering*, Reference [46].

16.10.2 Specific ALARP Assessment

Following the identification of gaps and risks, specific ALARP assessments have been carried out. This section describes possible improvement options to determine whether the options identified are reasonably practicable to be implemented in the design.

Improvement options are identified according to the requirements for optioneering and decision-making. Potential design changes with categories 1 and 2 are described in Table T-16.10-1. The analysis of other design changes is described in *ALARP Demonstration Report of PCSR Chapter 16, Reference [44] and Impact Analysis of Design Modification on Civil Engineering, Reference [46]*.

Once the reviews of improvement options discussed above have been completed, all of the identified areas for potential enhancement have been subject to optioneering and all of the reasonably practicable improvements have been incorporated, it is possible to demonstrate that the nuclear safety risks associated with the civil engineering works and structures are ALARP.

T-16.10-1 Improvement Process and Decision-making

No.	Improvement Process	Improvement Decision
1	A series of values of geotechnical parameters are	The geotechnical parameters are defined in *Generic Site Report* and *Generic Design*

No.	Improvement Process	Improvement Decision
	defined correlated to the declared shear wave velocity range in the *Generic Site Report*, including the ground stiffness and the allowable bearing capacity.	*Parameters for Civil Engineering, Reference [32]*. Based on these geotechnical parameters, a design change to the common raft foundation is undertaken. The thickness of the common raft is increased downwards and the steel reinforcement required is recalculated according to the increased thickness. The design change improves the design margin and the constructability.
2	The design reference (HPR1000 (FCG3)) only considers large passenger aircraft impact. According to a specific clause in the external hazard TAG, impact by other aircraft types should be considered as part of the GDA.	The large commercial aircraft impact is assessed and detailed information about the military aircraft impact analysis is documented in *Aircraft Impact Evaluation Method Statement, Reference [42] and Basis of Safety Case for Aircraft Impact, Reference [40]*.

16.10.3 ALARP Conclusion

For the civil engineering structures, following the holistic and specific ALARP assessment methodology and process described above, gaps have been identified and improvements have been carried out. It is demonstrated that the nuclear safety risks for the civil engineering area are ALARP for the GDA phase.

16.11 Concluding Remarks

This Chapter is one of the overall PCSR documents and presents the contents of the civil structures design to underpin the fundamental objective of the UK HPR1000 plant. The complete safety cases for the civil works and structures are established in this Chapter and relevant reference documents.

The civil engineering design of the UK HPR1000 has built upon the experience acquired from the reference design. An ALARP methodology and appropriate nuclear-specific codes and standards are applied in the analysis and design of civil engineering structures.

The LLSFs relevant to Civil Engineering are screened and identified. The safety functions and engineering requirements for the civil engineering structures are further

decomposed from the LLSFs. These requirements are recorded and satisfied in a series of documents.

The seismic analysis of civil engineering structures is carried out. The SSE-1 categorisation buildings are designed to remain essentially elastic under design basis earthquake loading, the SSE-2 classification buildings are designed to not affect the function of nuclear safety-related items and the non-nuclear seismic structures are designed in line with normal industrial codes and standards.

The design of nuclear safety-related structures is substantiated to be robust and achieve the engineering requirements under design basis condition as well as beyond design basis condition. The design of structures under design basis conditions includes the consideration of global stability, strength design, serviceability design, concrete barrier and non-barrier evaluation and durability design which are achieved based on codified design approaches. The evaluation of structures under beyond design basis condition includes ultimate capacity evaluation for internal containment, aircraft impact evaluation, and cliff-edge effects for extreme environmental hazards.

The constructability of buildings is considered and the designs of the civil structures can be safely constructed. In-service inspection and testing are provided to monitor the ageing and degradation effects on the civil structures and ensure that the levels of reliability and availability of the civil structures remain in accordance with the design over the lifetime of the NPP.

The detailed arguments and evidence that the acceptance criteria and engineering requirements have been satisfied are provided for the five sub-claims to ensure the safety case is valid, complete and evidential.

The design of the civil engineering structures is demonstrated to meet the requirements of the safety functions, produces a conservative design for GDA and ensures that the structures meet the design requirements over the design life.

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16.13 Appendix 1 Route Map of PCSR Chapter 16

T-16.13-1 Route Map of PCSR Chapter 16

Sub-claim	Argument	Evidence	Supporting Document
Sub-claim 3.3.9.SC16.1: The safety functional requirements have been derived for the civil works and structures.	**Argument 3.3.9.SC16.1-A1**: The safety functional requirements have been identified based on fundamental safety functions, described in Sub-chapter 16.3.1.	**Evidence 3.3.9.SC16.1-A1-E1**: The safety functional requirements of internal and external containment have been identified.	Basis of Safety Case for Internal and External Containment, Reference [4]
		Evidence 3.3.9.SC16.1-A1-E2: The safety functional requirements of the BRX internal structures have been identified.	Basis of Safety Case for BRX Internal Structures, Reference [5]
		Evidence 3.3.9.SC16.1-A1-E3: The safety functional requirements of the safeguard buildings have been identified.	Basis of Safety Case for Safeguard Buildings, Reference [6]
		Evidence 3.3.9.SC16.1-A1-E4: The safety functional requirements of the BFX have been identified.	Basis of Safety Case for BFX, GHXFXX10001DWJG42GN, Reference [7]
		Evidence 3.3.9.SC16.1-A1-E5: The safety functional requirements of the BNX, BWX, BDA/BDB/BDC/BDU/BDV, BEJ, BPX, and BEX have been identified.	Basis of Safety Case for BNX, Basis of Safety Case for BWX, Basis of Safety Case for Diesel Generator Buildings, Basis of

Sub-claim	Argument	Evidence	Supporting Document
			Safety Case for BEJ, Basis of Safety Case for BPX, Basis of Safety Case for BEX, Reference [8], [9], [10], [11], [12], and [13]
	Argument 3.3.9.SC16.1-A2: The classification and categorisation of the structures have been identified, described in Sub-chapter 16.3.2.	**Evidence 3.3.9.SC16.1-A2-E1**: The methodology of classification & categorisation has been developed.	Methodology of Safety Categorisation and Classification, Reference [47]
		Evidence 3.3.9.SC16.1-A2-E2: The classification and categorisation of civil structures have been identified based on the methodology.	Sub-chapter 16.3.2
	Argument 3.3.9.SC16.1-A3: The structure engineering requirements have been derived, described in Sub-chapter 16.3.3.	**Evidence 3.3.9.SC16.1-A3-E1**: The engineering requirements of internal and external containment have been identified.	Basis of Safety Case for Internal and External Containment, Reference [4]
		Evidence 3.3.9.SC16.1-A3-E2: The engineering requirements of the BRX internal structures have been identified.	Basis of Safety Case for BRX Internal Structures, Reference [5]
		Evidence 3.3.9.SC16.1-A3-E3: The engineering requirements of the safeguard	Basis of Safety Case for Safeguard

Sub-claim	Argument	Evidence	Supporting Document
		buildings have been identified.	Buildings, Reference [6]
		Evidence 3.3.9.SC16.1-A3-E4: The engineering requirements of the BFX have been identified.	Basis of Safety Case for BFX, GHXFXX10001DWJG42GN, Reference [7]
		Evidence 3.3.9.SC16.1-A3-E5: The engineering requirements of the BNX, BWX, BDA/BDB/BDC/BDU/BDV, BEJ, BPX, and BEX have been identified.	Basis of Safety Case for BNX, Basis of Safety Case for BWX, Basis of Safety Case for Diesel Generator Buildings, Basis of Safety Case for BEJ, Basis of Safety Case for BPX, Basis of Safety Case for BEX, Reference [8], [9], [10], [11], [12], and [13]
Sub-claim 3.3.9.SC16.2: The design of civil structures satisfies the safety functional requirements.	**Argument 3.3.9.SC16.2-A1**: Appropriate design codes and standards have been identified, described in Sub-chapter 16.4.	**Evidence 3.3.9.SC16.2-A1-E1**: The approach to codes and standards on civil structures has been developed.	The Approach to Codes and Standards for Civil Engineering, Reference [20]
		Evidence 3.3.9.SC16.2-A1-E2: Applicability of codes and standards on civil structures design has been developed.	Applicability of Codes and Standards for UK HPR1000, Reference [21]
	Argument 3.3.9.SC16.2-A2: The	**Evidence 3.3.9.SC16.2-A2-E1**: The	Structural Analysis and Design

Sub-claim	Argument	Evidence	Supporting Document
	methodology for civil structures analysis and design has been developed according to the codes and standards, described in Sub-chapter 16.5-16.7.	structural analysis and design method statement has been developed.	Method Statement, Reference [35]
		Evidence 3.3.9.SC16.2-A2-E2: Raft foundation analysis and design method statement has been developed.	Raft Foundation Analysis and Design Method Statement, Reference [36]
		Evidence 3.3.9.SC16.2-A2-E3: The internal containment analysis and design method statement has been developed.	Internal Containment Analysis and Design Method Statement, Reference [37]
		Evidence 3.3.9.SC16.2-A2-E4: The overall seismic analysis method statement has been developed.	Overall Seismic Analysis Method Statement, Reference [33]
		Evidence 3.3.9.SC16.2-A2-E5: The aircraft impact evaluation method statement has been developed.	Aircraft Impact Evaluation Method Statement, Reference [42]
	Argument 3.3.9.SC16.2-A3: Analysis and design of the structures have been conducted according to the methodology to ensure the safety	**Evidence 3.3.9.SC16.2-A3-E1**: Structural analysis and design, which include analysis and design of raft foundation and superstructure and reinforced concrete	Sub-chapter 16.6

Sub-claim	Argument	Evidence	Supporting Document
	functional requirements have been achieved, described in Sub-chapter 16.5-16.7.	barrier substantiation, have been developed.	
		Evidence 3.3.9.SC16.2-A3-E2: Seismic analysis for structures has been conducted.	Sub-chapter 16.5
		Evidence 3.3.9.SC16.2-A3-E3: Aircraft impact evaluation for structures has been conducted.	Sub-chapter 16.7.2
		Evidence 3.3.9.SC16.2-A3-E4: Ultimate capacity evaluation for internal containment has been conducted.	Sub-chapter 16.7.1
		Evidence 3.3.9.SC16.2-A3-E5: Cliff-edge effect of extreme environmental hazard for civil engineering structure has been conducted.	Sub-chapter 16.7.3
Sub-claim 3.3.9.SC16.3: All reasonably practicable measures have	**Argument 3.3.9.SC16.3-A1**: The civil structures meet the requirements of the relevant design principles and therefore of RGP, described in Sub-chapter 16.10.	**Evidence 3.3.9.SC16.3-A1-E1**: The consistency review against the RGP has been conducted.	Sub-chapter 16.10 ALARP Demonstration Report of PCSR Chapter 16, Reference [44]
		Evidence 3.3.9.SC16.3-A1-E2: The Gaps	ALARP Demonstration Report of

Sub-claim	Argument	Evidence	Supporting Document
been adopted to improve the design of the civil structures.		identified have been analysed and improved.	PCSR Chapter 16, Reference [44]
	Argument 3.3.9.SC16.3-A2: Design improvements have been considered and any reasonably practicable changes are implemented, described in Sub-chapter 16.10.	**Evidence 3.3.9.SC16.3-A2-E1**: The potential improvements against the RGP have been identified.	ALARP Demonstration Report of PCSR Chapter 16, Reference [44]
		Evidence 3.3.9.SC16.3-A2-E2: The potential improvements have been considered in the design of civil structures.	ALARP Demonstration Report of PCSR Chapter 16, Reference [44]
Sub-claim 3.3.9.SC16.4: The civil structures performance has been validated by commissioning and testing.	**Argument 3.3.9.SC16.4-A1**: The civil structures have been designed to take benefit from a suite of pre-construction tests, to provide assurance of the initial quality of the manufacture, described in Sub-chapter 16.8.	**Evidence 3.3.9.SC16.4-A1-E1**: The pre-construction tests for the materials have been conducted.	Sub-chapter 16.8
		Evidence 3.3.9.SC16.4-A1-E2: The pre-construction tests for the civil structure members have been conducted.	Sub-chapter 16.8
	Argument 3.3.9.SC16.4-A2: The civil structures have been designed to take benefit from a suite of commissioning tests, to provide assurance of the initial quality of the build, described in	**Evidence 3.3.9.SC16.4-A2-E1**: The tests on civil structures before commissioning have been conducted.	Sub-chapter 16.8

Sub-claim	Argument	Evidence	Supporting Document
	Sub-chapter 16.8.		
Sub-claim 3.3.9.SC16.5: The ageing effects of civil structures have been addressed in the design and suitable EMIT specified.	**Argument 3.3.9.SC16.5-A1**: The ageing effects have been considered in the design of the civil structures, described in Sub-chapter 16.9.	**Evidence 3.3.9.SC16.5-A1-E1**: The plant design life has been specified.	Sub-chapter 16.6.2
		Evidence 3.3.9.SC16.5-A1-E2: The design loads from environmental hazards are developed.	Sub-chapter 16.6.3
	Argument 3.3.9.SC16.5-A2: An initial EMIT strategy has been developed, described in Sub-chapter 16.9.	**Evidence 3.3.9.SC16.5-A2-E1**: The detailed approaches of building instrument and test have been developed.	Sub-chapter 16.9
		Evidence 3.3.9.SC16.5-A2-E2: The routine inspect and maintenance of buildings has been developed.	Sub-chapter 16.9

Note: The Sub-claim/Argument/Evidence code has been modified to present the link with the Overall Route Map. The current code "…x-Ay-Ez" of Claims, Arguments, and Evidences (CAE) stands for "Evidence x.y.z" in the previous CAE code. For example, Sub-claim 3.3.9.SC16.1 stands for Sub-claim 3.3.9.1 in the previous version, and Evidence 3.3.9.SC16.1-A1-E1 stands for Evidence 1.1.1.

16.14 Appendix 2 Screening of the LLSFs Relevant to Civil Engineering

T-16.14-1 **Screening of the LLSFs relevant to Civil Engineering for R - Reactivity Control**

Fundamental Safety Functions	High Level Safety Functions	Low Level Safety Functions (LLSFs)	Screening of the LLSFs relevant to Civil Engineering
R - Reactivity control	R1 - Maintain core reactivity control	R1-1. Control the boron concentration	The Civil Engineering is not related to the reactivity control functions.
		R1-2. Control the Reactor Coolant System (RCS) average temperature	
		R1-3. Manage rod configuration	
		R1-4. Prevent unacceptable core thermal power	
		R1-5. Prevent unacceptable rod configuration	
	R2 - Shutdown and maintain core sub-criticality	R2-1. Insert negative reactivity for core shutdown by rod insertion	
		R2-2. Insert negative reactivity for core shutdown by borated water injection	
		R2-3. Compensate for moderator effect during RCS cooling	
		R2-4. Ensure sufficient shutdown margin	
	R3 - Prevention of uncontrolled positive reactivity insertion into the core	R3-1. Protect the core from uncontrolled positive reactivity insertion by rods configuration	
		R3-2. Protect the RCS against dilution	
		R3-3. Protect RCS against overcooling	
	R4 - Maintain sufficient sub-criticality of fuel stored outside the RCS but within the site	R4-1. Maintain sufficient sub-criticality of fuel stored in fuel pool	
		R4-2. Maintain sufficient sub-criticality of spent fuels during operations and interim storage of spent fuels	

T-16.14-2 **Screening of the LLSFs relevant to Civil Engineering for H - Heat Removal**

Fundamental Safety Functions	High Level Safety Functions	Low Level Safety Functions	Screening of the LLSFs relevant to Civil Engineering
H - Heat removal	H1 - Maintain sufficient RCS water inventory for core cooling	H1-1. Ensure sufficient RCS inventory	The Civil Engineering is not related to the RCS water inventory maintaining functions.
		H1-2. Inject water into the RCS	
		H1-3. Prevent leaks from RCS through connected auxiliary lines	
		H1-4. Prevent leaks from RCS through main coolant pump seals	
		H1-5. Prevent leaks from RCS through high reliability equipment	
		H1-6. Limit leaks from RCS through connected auxiliary lines	
		H1-7. Limit leaks from RCS through main coolant pump seals	
		H1-8. Limit leaks from RCS through postulated break	
	H2 - Remove heat from the core to the reactor coolant	H2-1. Remove heat from the core by RCS forced flow	The Civil Engineering is not related to the RCS heat removal functions.
		H2-2. Remove heat from the core by RCS natural circulation	
		H2-3. Maintain the RCS pressure for heat removal	
		H2-4. Control the RCS pressure for heat removal	
	H3 - Transfer heat from the reactor coolant to the ultimate heat sink	H3-1. Reduce heat generation in RCS by limiting auxiliary thermal power source	
		H3-2. Remove heat by secondary side	
		H3-3. Remove heat through Residual Heat Removal system	
		H3-4. Remove heat through Containment Heat Removal System	
		H3-5. Remove residual heat by feed and bleed	
		H3-6. Remove heat by in-vessel retention injection	

Fundamental Safety Functions	High Level Safety Functions	Low Level Safety Functions	Screening of the LLSFs relevant to Civil Engineering
	H4 - Maintain heat removal from fuel stored outside the RCS but within the site	H4-1. Prevent the fuel pool from leakage	The “prevent” function is required to perform before the fault or accident occurring, and the “protect” function is required to perform after the fault or accident occurring, which belong to different level of Defence in Depth. The prevention of the fuel pool leakage is realised by the robust design (e.g. SSC classification and temperature control). They are not related to the Civil Engineering.
		H4-2. Protect the fuel pool from leakage	F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release.（BFX SFP）
		H4-3. Remove heat from the fuel pool by Fuel Pool Cooling and Treatment System	Not related to the Civil Engineering.
		H4-4. Remove the decay heat by natural ventilation during interim storage period	Not related to the Civil Engineering.

T-16.14-3 **Screening of the LLSFs relevant to Civil Engineering for C - Confinement**

Fundamental Safety Functions	High Level Safety Functions	Low Level Safety Functions	Screening of the LLSFs relevant to Civil Engineering
C - Confinement	C1 - Maintain integrity of the fuel cladding to ensure confinement of radioactive material	C1-1. Control the core power distributions (linear power density & axial offset)	Not related to the Civil Engineering.
		C1-2. Prevent unacceptable core power distributions (linear power density & axial offset)	
		C1-3. Prevent pellet clad interaction	
	C2 - Maintain integrity of the Reactor Coolant Pressure Boundary to ensure confinement of radioactive material	C2-1. Ensure Reactor Coolant Pressure Boundary (RCPB) leak tightness	
		C2-2. Prevent primary water release from RCS connected auxiliary lines	
		C2-3. Protect against primary water release from RCS connected auxiliary lines	
		C2-3. Prevent RCS Pressurised Thermal Shock	
		C2-4. Prevent the pressurizer filling-up	
		C2-5. Protect the RCS against overpressure	
	C3 – Maintain integrity of reactor containment to ensure confinement of radioactive material	C3-1. Extend containment to the secondary system and auxiliary systems connected to RCS	Not related to the Civil Engineering.
		C3-2. Maintain containment building structural integrity	F1: Function to provide structural support to SSC (Internal Containment/External Containment)
		C3-3. Maintain containment building leak tightness	F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release. (Internal

Fundamental Safety Functions	High Level Safety Functions	Low Level Safety Functions	Screening of the LLSFs relevant to Civil Engineering
			Containment /External Containment/Common raft (BRX part))
	C4 – Maintain integrity of the fuel stored outside of reactor containment	C4-1. Maintain integrity of the fuel stored in the fuel pool	The integrity is for the fuel assemblages during operations and storage. Not related to the Civil Engineering.
		C4-2. Maintain the radioactive confinement and the integrity of spent fuels during operations and interim storage	
	C5 - Store the radioactive material	C5-1. Store the radioactive material as gaseous waste	F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release. (For the buildings store the radioactive material as gaseous waste)
		C5-2. Store the radioactive material as liquid wastes	F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release. (BFX Pools except SFP, other building or part of the building which store the radioactive material as liquid wastes)
		C5-3. Store the radioactive material as solid wastes	F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release. (For the buildings store the radioactive material as solid wastes)
	C6 - Shield against radiation, control planned radioactive releases, and limit accidental radioactive releases	C6-1. Shield against radiation	F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release.
		C6-2. Control planned radioactive releases	Not related to the Civil Engineering.
		C6-3. Limit accidental radioactive releases	F4: Function to confine the radioactive materials, shield radiation, and reduce radioactive release. (Not related to the containment, C3-2 and C3-3 are required for containment)

T-16.14-4 **Screening of the LLSFs relevant to Civil Engineering for E - Extra Functions**

Fundamental Safety Functions	High Level Safety Functions	Low Level Safety Functions	Screening of the LLSFs relevant to Civil Engineering
E - Extra functions	E1 - Support Type R, H or C safety function	E1-1. Provide necessary plant information for the performance of safety functions	Not related to the Civil Engineering.
		E1-2. Provide mechanical supporting for the performance of safety functions	
		E1-3. Provide electric supporting for the performance of safety functions	
		E1-4. Provide other supporting for the performance of safety functions	F1: Function to provide structural support to SSC (all structures and foundation rafts)
	E2 - Prevent, protect and mitigate hazards impact	E2-1. Prevent internal hazards risk	The “prevent” function is required to perform before the fault or accident occurring, and the “protect” function is required to perform after the fault or accident occurring, which belong to different level of Defence in Depth. The preventions of hazards are realised by the robust design (e.g. appropriate SSC classification, elimination of combustible inventory by adoption of suitable materials. They are not related to the Civil Engineering.
		E2-2. Prevent external hazards risk	

Fundamental Safety Functions	High Level Safety Functions	Low Level Safety Functions	Screening of the LLSFs relevant to Civil Engineering
		E2-3. Protect internal hazards risk	F3: Function to provide protection to SSC against internal hazards. (All barriers identified by the Internal Hazards discipline)
		E2-4. Protect external hazards risk	F2: Function to provide protection to SSC against external hazards. (The external walls and roofs, foundation rafts, and external containment)
		E2-5. Mitigate internal hazards risk	The mitigation of hazards are realised by design measures such as the fire sprinklers. Not related to Civil Engineering.
		E2-6. Mitigate external hazards risk	

Note: the “prevent” function is required to perform before the fault or accident occurring, and the “protect” function is required to perform after the fault or accident occurring, which belong to different levels of Defence in Depth.