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Table of Contents

20.1 Introduction 4
20.1.1 Purpose and Scope 4
20.1.2 Assumptions 5
20.1.3 Interfaces with other SSEC Chapters 5
20.2 Description of Civil Engineering SSCs 7
20.3 Civil Engineering Claims, Arguments, Evidence 13
20.4 Codes, Standards and Methodology 15
20.4.1 Codes, Standards and Methodologies used for the Civil Structures of SMR-300 15
20.4.2 UK and International Guidance used in Development of the Generic SMR-300 16
20.4.3 CAE Summary 18
20.5 Design of Civil SSCs 19
20.5.1 Safety Functions of Civil Structures 19
20.5.2 SMR Class and Seismic Category of Civil Structures and SSCs 20
20.5.3 Design Codes and Standards Applied to Civil SSCs 21
20.5.4 Analysis and Design Methodology 22
20.5.5 CAE Summary 28
20.6 Defence in Depth 29
20.6.1 Conservative Design 29
20.6.2 Design Basis Analysis 29
20.6.3 Beyond Design Basis Accident Analysis 29
20.6.4 CAE Summary 30
20.7 Quality Manufacturing And Installation 31
20.7.1 Quality Assurance 31
20.7.2 Construction/Fabrication 31
20.7.3 Examination, Inspection, Maintenance and Testing (EIMT) 32
20.7.4 CAE Summary 32
20.8 Chapter Summary and Contribution to ALARP 33
20.8.1 Technical Summary 33
20.8.2 ALARP Summary 34
20.8.3 Conclusion 36
20.9 References 37
20.10 List of Appendices 41
Appendix A CAE Route Map A-1
Appendix B High-Level, Safety and Non-Safety Functions for Civil Structures B-1

List of Figures

Figure 1: SMR-300 Site Layout 7

Figure 2: Plan Arrangement of Nuclear Island Structures 7

Figure 3: Arrangement of the Containment Structure, Annular Reservoir and Containment Enclosure Structure 8

Figure 4: Typical Configuration of CES SC Modules 8

Figure 5: Plan of the Intermediate Building at Below Grade Elevation 10

List of Tables

Table 1: Claims and PSR Subchapters 14

Table 2: US NRC Regulations and Regulatory Guidance used for the Civil Structures of SMR-300 16

Table 3: Principal Codes and Standards used for the Civil Structures of SMR-300 16

Table 4: UK RGP for Civil Engineering 17

Table 5: High-Level Plant Functions for Civil Engineering SSCs 19

Table 6: SMR Class and Seismic Categories for Civil Engineering SSCs 21

Table 7: Principal Codes and Standards applied to Civil Engineering SSCs 21

Table 8: Containment Structure Loads 22

Table 9: Seismic Category I Concrete Structures Loads 23

Table 10: Seismic Category I (other than the CS) Steel Structures Loads 24

Table 11: Seismic Category I Steel-Plate Composite Structure Loads 25

Table 12: PSR Part B Chapter 20 CAE Route Map A-1

20.1 INTRODUCTION

The Fundamental Purpose of the Generic Design Assessment (GDA) Safety, Security and Environment Case (SSEC) is to demonstrate that the Generic Small Modular Reactor (SMR)-300 can be constructed, operated, and decommissioned on a generic site in the UK to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment as defined in the Preliminary Safety Report (PSR) Part A Chapter 1 Introduction [1].

The Fundamental Purpose is achieved through the Fundamental Objective of the PSR, which is to summarise the safety standards and criteria, safety management and organisation, claims, arguments and intended evidence to demonstrate that the Generic SMR-300 design risks to people are likely to be tolerable and As Low as Reasonably Practicable (ALARP) [1] .

PSR Part B Chapter 20 presents the Claims, Arguments and intended Evidence (CAE) for the design of Civil Engineering Structures, Systems and Components (SSCs) that underpin the design of the Generic SMR-300.

20.1.1 Purpose and Scope

The Overarching SSEC Claims are presented in PSR Part A Chapter 3 Claims, Arguments & Evidence [2].

This chapter (PSR Part B Chapter 20) links to the overarching claim through Claim 2.2:

Claim 2.2: The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

As set out in PSR Part A Chapter 3, Claim 2.2 is further decomposed across several engineering disciplines which are responsible for development of the design of relevant SSCs. This chapter presents the civil engineering aspects for the generic SMR-300 and therefore directly supports a claim focused on the overall design and architecture of civil structures, Claim 2.2.11.

Claim 2.2.11: The overall design and architecture of Civil SSCs ensure that safety functions and non-safety functions are delivered and faults arising from failures of the SSCs are minimised.

Further discussion on how the Level 3 claim is broken down into Level 4 claims and how the Level 4 claims are met is provided in subchapter 20.3.

The Civil Engineering SSCs included within the scope of this chapter are those within the Nuclear Island (NI) (refer to the GDA Scope Report [3]):

- The Containment Structure (CS)
- The Containment Enclosure Structure (CES)
- The Annular Reservoir (AR), as the inner and outer walls of the AR are formed by the CS and CES, respectively
- The Reactor Auxiliary Building (RAB)
- The Intermediate Building (IB)
- The Radioactive Waste Building (RWB)

This chapter covers the codes and standards associated with the design of these SSCs (subchapter 20.4), analysis methodologies (subchapter 20.5), the defence-in-depth associated with the design of the SSCs (subchapter 20.6) and the quality manufacturing and installation approach (subchapter 20.7). Finally, a summary of considerations against the ALARP principle is provided, together with any forward actions or commitments that have arisen (subchapter 20.8).

There are three aspects of the Civil Engineering design which are novel, with respect to their application in the UK specifically: a significant portion of the reactor building is partially embedded below grade; the design of the CES adopts Steel-Concrete Composite (SC) modular construction and the design basis and testing are in development; furthermore, the AR is formed by the inner and outer walls of the CES and CS, respectively.

The Interim Spent Fuel Storage Installation (ISFSI) is excluded from the scope of this chapter as its design development is still at an early stage [3]. Furthermore, internal structures and equipment supports are excluded from the scope of this chapter. Structures that are not in the scope of the GDA but contain equipment which perform safety functions will be addressed in future safety reports.

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in PSR Part A Chapter 2 General Design Aspects and Site Characteristics [4].

20.1.2 Assumptions

No assumptions are identified in this revision.

20.1.3 Interfaces with other SSEC Chapters

The Civil Engineering discipline interfaces with multiple topic areas across the plant.

The general design aspects and site characteristics to be used for the design and analysis of Civil Engineering SSCs are reported in PSR Part A Chapter 2 General Design Aspects and Site Characteristics [4]. PSR Part A Chapter 2 also introduces the Civil Engineering SSCs of the SMR-300.

PSR Part B Chapter 1 Reactor Design and Engineering Aspects [5], PSR Part B Chapter 2 Reactor Fuel and Core [6] and PSR Part B Chapter 5 Description of the Reactor Supporting Facilities [7] provide a description of the SSCs contained within the Civil Engineering SSCs of the SMR-300.

The Civil Engineering discipline also interfaces with PSR Part B Chapter 9 Description of Operational Aspects/Conduct of Operations [8] (which provides a description of the relevant Examination, Inspection, Maintenance and Testing (EIMT) arrangements for civil structures and states the limits and conditions of the safety case), PSR Part B Chapter 25 Construction and Commissioning Approach [9] (for the methods of construction of the Civil Engineering SSCs) and PSR Part B Chapter 26 Decommissioning Approach [10] (for the methods of dismantling and decommissioning of the Civil Engineering SSCs).

PSR Part B Chapter 10 Radiological Protection [11], PSR Part B Chapter 14 Design Basis Accident Analysis [12], PSR Part B Chapter 15 Beyond Design Basis and Severe Accident Analysis, and Emergency Preparedness [13], PSR Part B Chapter 16 Probabilistic Safety Analysis [14] will provide a comparison against risk targets defined in PSR Part B Chapter 2

Reactor Fuel and Core [6]. PSR Part A Chapter 5 Summary of ALARP [15] concludes that it can be demonstrated that the Generic SMR-300 reduces risks to ALARP and that the Fundamental Purpose of the SSEC has been fulfilled.

20.2 DESCRIPTION OF CIVIL ENGINEERING SSCs

The following provides a summary description of the Civil Engineering SSCs that are within the scope of this PSR chapter.

The generic site layout of the SMR-300 is illustrated in Figure 1. A plan layout of the NI structures is given in Figure 2.

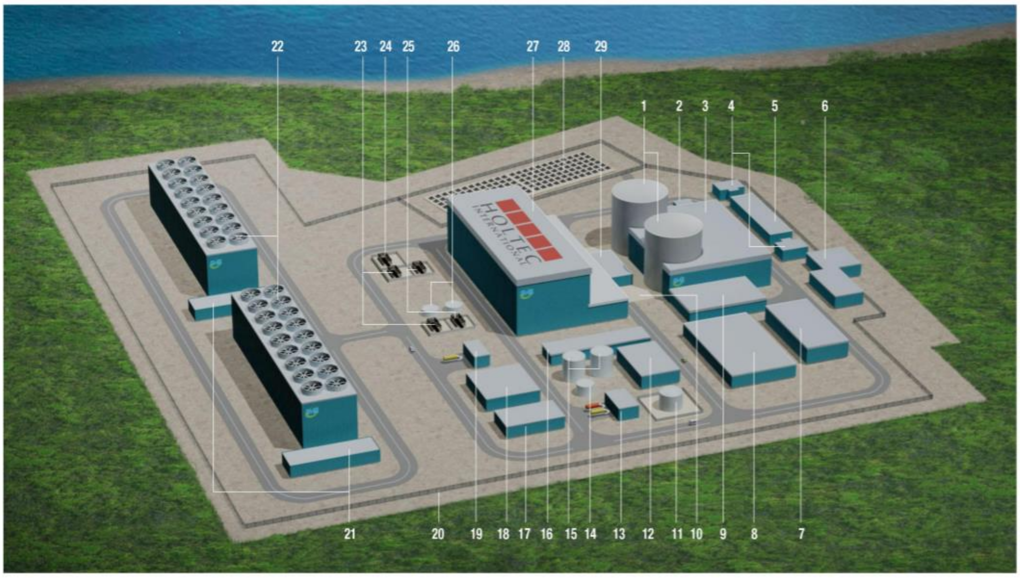

Figure 1: SMR-300 Site Layout

Figure 2: (REDACTED)

PSR Part A Chapter 2 General Design Aspects and Site Characteristics [4] gives the site layout and main buildings description. The main Civil Engineering SSCs included within the NI are:

- The CES (number 1 on Figure 1) and the incorporated CS and AR are described within subchapters 20.2.1.1, 20.2.1.2 and 20.2.1.3, respectively.
- The IB (number 10 on Figure 1) is described within subchapter 20.2.1.4.
- The RAB (number 3 on Figure 1) is described within subchapter 20.2.1.5.
- The RWB (number 9 on Figure 1) is described within subchapter 20.2.1.6.

These structures are described separately within this subchapter.

As explained in subchapter 20.1, the ISFSI is excluded from the scope of this chapter and it is dealt within PSR Part B Chapter 24 Fuel Transport and Storage [16].

20.2.1.1 Containment Enclosure Structure

The CES consists of a cylindrical structure that fully envelops the CS (Figure 3). It is a large deeply founded structure designed for the following functions:

- To protect the CS from external hazards and threats.
- To provide shielding to the plant and personnel from radioactive sources inside the CS during power operations and postulated accidents.
- To form the outer wall of the AR; the AR water inventory provides passive containment heat removal during a Loss-Of-Coolant Accident (LOCA).
- To provide a vent for the AR to facilitate evaporative cooling.
- The CES interfaces with the IB, which protects the main steam and main feedwater lines to the point of their respective safety isolations and seismic restraints. The CES provides support for these lines at the CS penetration.

The CES is constructed with modular SC walls for the above and below grade portions. It shares a common RC basemat with the CS. Two concentric steel shells form the inner and outer faces of the SC modules, with interconnecting plates providing support (see Figure 4). Each section is shop fabricated and transported to the site where it is welded to adjacent sections to form rings, which are stacked and filled with concrete.

Containment penetrations and personnel access to the containment are made via openings in the below-grade section of the CES. An equipment penetration is located at ground level to facilitate replacement of major components and for access during a refuelling or maintenance outage.

At the top, the CES presents a lid with openings for the vent pipes and vertical access for steam generator (SGE) removal, if such replacement becomes necessary. These vents allow water vapor from the AR to escape during a postulated design basis accident. The lid is hardened against Design Basis Threats (DBTs) and other objects such as external missiles.

All surfaces of the CES in contact with the stored water in the AR are coated with a protective coating for corrosion resistance.

Figure 3: (REDACTED)

Figure 4: (REDACTED)

20.2.1.2 Containment Structure

The CS is a cylindrical steel containment vessel with a domed upper head and a steel-lined reinforced concrete base. It is partially embedded below grade. Figure 3 illustrates the arrangement of the CS.

The primary functions of the CS are to:

- Provide a leak-tight barrier to contain fission product releases from the reactor coolant pressure boundary during operational states and design basis events.
- Contain the mass and energy release from a postulated LOCA and secondary-system pipe ruptures.
- Contain and support the Reactor Pressure Vessel (RPV), Reactor Coolant System (RCS), Spent Fuel Pool (SFP), and associated SSCs.

The CS houses the following systems:

- RCS
- Passive core cooling system (PCC) which consists of:
    - Primary decay heat removal system (PDHR)
    - Secondary decay heat removal system (SDHR)
    - Automatic depressurization system (ADS)
    - Passive core makeup water system (PCM)
- Containment building ventilation system (CBV)
- Combustible gas control system (CGC)
- SFP
- Light load handling system (LLH)
- Overhead heavy load handling system (CSH)

The CS is enveloped and protected from external hazards and threats by the CES. The CS and the CES share a common basemat. Furthermore, the CS forms the inner wall of the AR which contains a substantial captive body of water and has the primary function to provide a heat sink in the event of a Design Basis Accident (DBA).

The CS is fabricated of low-alloy carbon steel. The lateral part of the CS above grade is reinforced with internally mounted stiffener rings.

The CS is designed to have a clear path above the SGE to the domed upper head; the SGE is aligned with the CES vent to facilitate replacement when needed and to support equipment removal during decommissioning. For most refuelling and maintenance activities, access to the CS is through the ground-level containment equipment hatch. The CS equipment hatch is a bolted and gasketed round hatchway, providing access to the CS interior from the RAB. A personnel airlock provides an additional entrance during outages and power operations.

The CS features a polar crane, supported by the CS shell. The crane is equipped with a primary hook and an auxiliary hook. Floor equipment hatches are provided at each floor elevation to allow for vertical movement of equipment. The floor equipment hatches are aligned such that the polar crane can lift equipment directly from the lowest elevation. The crane access and capacity are designed for refuelling operations, including handling of the reactor internals and equipment needed for dry storage of spent fuel.

A bridge crane located in the refuelling/operating deck is used for fuel movements in and between the SFP and the RPV. The bridge spans the SFP and runs on rails set into the edge of the SFP. The crane is equipped with a fuel mast for fuel movement and an auxiliary hoist for non-fuel related operations in the pool.

20.2.1.3 Annular Reservoir

The AR is contained in an approximately 2.4m wide annulus between the CS and CES; the CS and CES form the inside and outside wall of the AR, respectively. Below grade, the space between the CS and CES is filled in with lean concrete. A watertight seal is provided at grade level between the CS and CES to form the bottom of the AR.

The AR is a component of the Passive Containment Heat Removal System (PCHR). The AR contains a substantial captive body of water, and its primary function is to provide the heat sink of the PDHR and SDHR in the event of a DBA. The AR has sufficient capacity to accept heat from the core, spent fuel pool, and containment during a DBA.

Heat from the core may be transferred from the SGE to the AR through the SDHR heat exchanger (HX). The SDHR HX is a part of a passive safety system, with natural circulation driving flow through the heat exchanger. Additionally, heat from the CS atmosphere may be transferred to the AR by conduction through the steel CS wall. Heat is rejected to the atmosphere by evaporation through the CES top vent.

A monorail supported by the CES inner wall allows for equipment movement within the AR for inspection or repair. Additionally, there are provisions to detect leaks from the AR. The AR contains a raised section which allows the IB to interface with the CS. The IB contains main steam and main feedwater piping and connects the CS to the Turbine Building.

The AR ensures leak-tightness by the provision of a steel cover that rests on the concrete infill and is welded to the CS shell and CES inner wall faceplate.

20.2.1.4 Intermediate Building

The IB is an RC structure which interfaces with the CES and protects the Main Steam System (MSS) and Main Feedwater System (MFS) lines following their exit from the CES and before the Turbine Building (TB). The IB houses the main steam and main feedwater lines, as well as the Atmospheric Dump Valves (ADVs), Safety Relief Valves (SRVs) and the Main Steam Isolation Valves (MSIV).

The CES provides support for these lines at the CS penetration. The AR contains a raised section which allows the IB to interface with the CS. The IB connects the CS to the TB.

The arrangement of the IB is illustrated in Figure 2 and a plan view is shown in Figure 5.

Figure 5: (REDACTED)

20.2.1.5 Reactor Auxiliary Building

The RAB is a partially embedded structure that houses mechanical and electrical equipment used for normal plant operation and safe shutdown of the reactor. The RAB combines the functions of an auxiliary building and a traditional fuel handling building into a singular building. The plan arrangement of the RAB is illustrated in Figure 2.

The RAB is a three-level building with one level at grade and two levels below grade. It is constructed mostly from RC. There are two main sections of the RAB. The non-radiologically controlled area houses the Main Control Room (MCR) and the majority of the electrical equipment and switchgear. The Radiologically Controlled Area (RCA) contains systems to support normal primary plant operations. The RAB is located next to the CES.

The RAB is designed to perform the following functions:

- To provide protection and separation for the seismically designed mechanical and electrical equipment located outside the containment.
- To house SSCs that are critical for the normal operation of the plant.

The RCA contains the systems and equipment that are or may be radioactive or contaminated. The RCA extends to all three levels of the RAB, including the Fuel Handling Area (FHA) on the grade floor and below grade elevations housing plant auxiliary systems.

The major equipment, systems and functions contained within the RCA are:

- Remote Shutdown Facility (RSF)
- Main equipment hatch and personnel hatch for entry to the CS
- Residual heat removal system (RHR)
- Chemical and volume control system (CVC)
- Mechanical equipment rooms
- Containment mechanical piping penetration areas
- Containment electrical penetration areas
- FHA
- Chilled water system (CWS) and select ventilation systems
- SFP cooling system
- Some radiological waste handling equipment and functions.

The FHA has:

- An area for receiving and inspecting new fuel assemblies.
- An area for storing new fuel assemblies.
- An area for loading new fuel assemblies into a HI-TRAC transfer cask which is the transported into the CS.
- An area to prepare spent nuclear fuel for dry storage.

The FHA features a bridge crane to support new fuel transfer activities. One bay of the FHA is allocated to receive new fuel from trailers or rail cars. Space is provided for inspection of the assemblies. The FHA includes provisions to use a low-profile transporter to move transfer casks in and out of the CS through the main equipment hatch. A separate bay is provided to prepare spent fuel for dry storage. Spent fuel is stored in the SFP inside the CS, then transferred to the FHA for processing in preparation for onsite interim dry storage using the HI-STORM UMAX system. Spent fuel is not stored in the FHA.

The non-radiologically controlled portion of the RAB, in accordance with the US Nuclear Regulatory Commission (US NRC) specifications, is contained in the two upper levels of the RAB (the ground level and the first of the two below-grade levels), adjacent to the RCA. Its primary purposes are to support the Main Control Room (MCR) and house the safety-related electrical systems.

The non-radiologically controlled area of the RAB contains the control point (change room) into the RCA. The control point is located on the ground level and is provided with decontamination equipment. Safety-related instrumentation and control cabinets, motor controllers, switchgear, and battery banks are housed on both levels. The control rod drive system electrical equipment is located on the lower level. The two divisions of safety-related electrical equipment are separated by elevation; one division is located on the lower level, while the other is located on the ground level.

The lower level of the non-radiologically controlled portion of the RAB houses the Control Room Emergency Zone (CREZ), including the MCR and the MCR habitability system (MCH). The MCR is designed to provide a habitable area from which to safely operate the reactor. A kitchen, restroom facilities, conference room, and office are also provided adjacent to the MCR within the non-radiologically controlled portion of the RAB.

The MCR's location below grade improves security and provides better protection from external hazards. The RSF is located on the same level of the RAB as the MCR. Fire doors connecting the RCA and the non-radiologically controlled area of the RAB provide a pathway for rapid access to the RSF if the MCR becomes uninhabitable.

20.2.1.6 Radioactive Waste Building

The RWB (Figure 2) is an RC structure. It accommodates areas for processing and segregating waste for storage or shipping.

The Radioactive Waste Building HVAC System (RBV), the Radioactive Drain System (RDS) and the Solid Radwaste System (SRW) are contained within the RWB.

The RWB is a non-seismic structure designed to comply with the seismic requirements in Regulatory Guide (RG) 1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants [17].

20.3 CIVIL ENGINEERING CLAIMS, ARGUMENTS, EVIDENCE

The primary purpose of a CAE approach is to capture the golden thread of a safety case narrative to demonstrate how plant and operational evidence is brought together and to justify that a high-level or fundamental claim is true. In the context of the GDA of the generic SMR-300, that is how the Fundamental Purpose of the overarching SSEC (presented in PSR Part A Chapter 1 Introduction [1]) is achieved.

The Fundamental Purpose follows a golden thread throughout the SSEC to CAE via the objectives of the PSR, the Preliminary Environmental Report (PER) and the Generic Security Report (GSR). The overarching SSEC claims and the philosophy for their architecture is presented in PSR Part A Chapter 3 Claims, Arguments & Evidence [2].

This chapter contributes directly to Claim 2.2, which is focused on the demonstration of the design and that the SSCs that form the design, are developed to ensure they meet the relevant safety requirements and appropriate codes and standards.

Claim 2.2: The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

As set out in PSR Part A Chapter 3, Claim 2.2 is further decomposed across several engineering disciplines which are responsible for development of the design of relevant SSCs. This chapter presents the civil engineering aspects for the generic SMR-300 and therefore directly supports a claim focused on the overall design and architecture of civil structures, Claim 2.2.11.

Claim 2.2.11: The overall design and architecture of Civil structures ensure that safety functions and non-safety functions are delivered and faults arising from failures of structures are minimised.

Claim 2.2.11 has been further decomposed within PSR Part B Chapter 20, across the design lifecycle, to provide confidence that the relevant requirements on structures will be met during all lifecycle phases.

This has been done by breaking down Claim 2.2.11 into four further sub-claims.

Claim 2.2.11.1 contributes to the *Design* phase by defining the codes and standards that the design will be assessed against. Claim 2.2.11.2 is also important to the *Design* phase, by ensuring that civil structures are analysed using best practice engineering analysis methodologies.

Claim 2.2.11.3 is important to the *Safety Analyses* phase, by ensuring that defence in depth is provided at multiple independent levels, so that the failure of one of those levels is accommodated by other engineered safety features within the design.

Claim 2.2.11.4 then ensures civil structures achieve their design intent through quality *Manufacturing* and *Installation* processes, noting that the maturity of evidence for this claim will be limited at a PSR stage. Claim 2.2.11.4 also covers through-life *Operational* maintenance aspects for civil structures noting the overall approach to EIMT is provided in PSR Part B Chapter 9 Description of Operational Aspects and Conduct of Operations [8].

Table 1 shows in which subchapters of this PSR chapter these claims are demonstrated to be met.

Table 1: Claims and PSR Subchapters

Claim No	Claim	Subchapter
2.2.11.1	Civil SSCs are designed using appropriate Codes and Standards, taking cognisance of RGP and OPEX.	20.4 CODES, STANDARDS AND METHODOLOGIES
2.2.11.2	Civil SSCs are designed using best practice analysis methodologies, taking cognisance of RGP and OPEX.	20.5 DESIGN OF CIVIL SSCs
2.2.11.3	Defence in depth of Civil SSCs is maintained so far as is reasonably practicable through good engineering practice.	20.6 DEFENCE IN DEPTH
2.2.11.4	Civil SSCs achieve the design intent through quality manufacturing and installation process.	20.7 QUALITY MANUFACTURING AND INSTALLATION

A summary of the current CAE route map for PSR Part B Chapter 20 is provided in Appendix A which is taken from the Generic SMR-300 Overarching SSEC Claim Routemap presented in Appendix A of PSR Part A Chapter 3. A further update on claim decomposition, argument development and evidence maturity will be provided in the subsequent update of the chapter.

20.4 CODES, STANDARDS AND METHODOLOGY

Claim 2.2.11.1: Civil SSCs are designed using appropriate Codes and Standards, taking cognisance of RGP and OPEX.

This subchapter outlines the codes and standards used in the design of SMR-300 Civil Structures.

The Requesting Party (RP) has recognised that UK nuclear safety regulations are based on a non-prescriptive regime and consequently the technical codes and standards that must be used for nuclear power plant are not prescribed. However, the codes and standards must represent RGP. New codes and standards can be introduced where needed by say, novel design features. Use of such codes will be justified in each case.

20.4.1 Codes, Standards and Methodologies used for the Civil Structures of SMR-300

The SMR-300 is designed in accordance with US NRC regulatory guidance and applicable US Code of Federal Regulations (CFR). The standard design of the SMR-300 intends to comply with US NRC requirements set forth in Title 10 CFR Part 50 [18].

The selection of codes and standards applied to the development and design of the SMR-300 is commensurate with the importance of the relevant safety functions delivered. The selection of codes and standards is imparted from the classification of SSCs based on US NRC requirements. SSCs of the SMR-300 are classified according to their importance to safety. The classification assigned determines requirements for design and construction of the SSCs, including compliance to codes or standards.

The applicable US NRC regulations and RGs for the civil structures of the SMR-300 are summarised in Table 2, whereas the principal codes and standards used for the design of civil structures for the SMR-300 standard design are summarised in Table 3. This is further elaborated upon in subchapter 20.5.

The principal codes and standards applied to the design of nuclear safety related Civil Engineering SSCs are nuclear-specific and are considered RGP by the UK nuclear industry. More information is provided in subchapter 20.4.2. Any significant gaps in their application in the UK context will be identified and included in Revision 1 of this chapter.

Table 2: US NRC Regulations and Regulatory Guidance used for the Civil Structures of SMR-300

Label	Title	Revision/ Date
US NRC 10 CFR Part 50	Domestic Licensing of Production and Utilization Facilities [18]	-
US NRC 10 CFR Part 50, Appendix A	General Design Criteria for Nuclear Power Plants [19]	-
US NRC 10 CFR Part 50, Appendix S	Earthquake Engineering Criteria for Nuclear Power Plants [20]	
US NRC 10 CFR Part 100	Reactor Site Criteria [21]	-
US NRC RG 1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containment) [22].	2020
US NRC RG 1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants, U.S. Nuclear Regulatory Commission' [17]	2001
US NRC RG 1.29	Seismic Design Classification for Nuclear Power Plants' [23].	2021
US NRC RG 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components' [24]	2013
US NRC RG 1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants' [25]	2014
US NRC NUREG-0800, SRP 3.3.2	Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: Design of Structures, Components, Equipment, and Systems - Tornado Loadings [26]	2007
US NRC NUREG-0800, SRP 3.8.2	Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: Design of Structures, Components, Equipment, and Systems - Steel Containment [27].	2010
US NRC NUREG-0800, SRP 3.8.4	Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: Design of Structures, Components, Equipment, and Systems - Other Seismic Category I Structures [28].	2013

Table 3: Principal Codes and Standards used for the Civil Structures of SMR-300

Label	Title	Revision/ Date
ACI 349-13	Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary [29]	2014
ANSI/AISC N690-18, Appendix N9	Specification for Safety-Related Steel Structures for Nuclear Facilities [30]	2018
ASCE/SEI 4-16	Seismic Analysis of Safety-Related Nuclear Structures' [31]	2017
ASME BPVC.III.1.NE	Boiler and Pressure Vessel Code (BPVC) Section III - Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NE – Class MC Components [32]	2023
ASME BPVC.III.NCA	Boiler and Pressure Vessel Code (BPVC) Section III – Rules for Construction of Nuclear Facility Components – Subsection NCA – General Requirements for Division 1 and Division 2 [33]	2023

20.4.2 UK and International Guidance used in Development of the Generic SMR-300

No novel or in-house (bespoke) design codes have been applied in the development and design of the SMR-300. The principal codes and standards applied to the design of nuclear safety related Civil Engineering SSCs are nuclear-specific and are considered RGP by the UK nuclear industry. This is based on existing practices adopted on UK nuclear licensed sites, application in earlier and successful GDAs, as well as recognition as RGP by Office for Nuclear Regulation (ONR) Technical Assessment Guides (TAGs). For example, ASCE/SEI 4 Seismic

Analysis of Safety-Related Nuclear Structures [31] and ACI 349 Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary [29] are indicated as RGP for seismic analysis of nuclear related civil engineering structures and blast/impact loading respectively in NS-TAST-GD-017 ONR Guide - Civil Engineering, Nuclear Safety Technical Assessment Guide [34]. No further IAEA or WENRA documentation has been considered.

Several of the principal codes and standards have no equivalent British codes and standards. Examples are ACI 349 [29], ANSI/AISC N690 [30] and ASCE/SEI 4 [31].

The applicable UK RGP is therefore the same as in Table 3 for the SMR-300 Standard Design. Table 4 clarifies the UK documentation considered.

Table 4: UK RGP for Civil Engineering

Label	Title	Revision/ Date
ONR Documentation		
NS-TAST-GD-017	ONR Nuclear Safety Technical Assessment Guide (TAG) - Civil Engineering [34]	2022
ONR SAPs	ONR Safety Assessment Principles (SAPs) for Nuclear Facilities [35]	2020
IAEA Documentation		
N/A	Not considered further	-
WENRA Guidance		
N/A	Not considered further	-
‘Other’ Guidance		
N/A	Not considered further	-

20.4.2.1 Lessons Learnt

To demonstrate understanding of ONR’s expectations for the PSR, a review of the Regulatory Observations (ROs)/Regulatory Issues (RIs) relevant to Civil Engineering from previous GDAs, ONR Generic Design Assessment - Assessment of Reactors [36], has been undertaken to make sure that RGP, OPEX and important lessons learnt are considered at this stage.

A summary of the common lesson learnt related to Civil Engineering is provided below from ONR-GDA-GD-007 New Nuclear Power Plants: Generic Design Assessment Technical Guidance [37]:

- Robust categorisation and classification considering the effects on SSCs is needed. Better demonstration of design basis analysis and cliff edge effects is needed. The Requesting Party (RP) should consider the effect of qualification of monitoring equipment and Class 1 barriers on civil engineering structures. [Will be considered in response to relevant safety classification and standards SAPs]
- It is considered good practice that the seismic models for SSI and Structural Analysis include FE Models. [The design of the SMR-300 adopts state-of-the-art SSI and Structural Analysis FE models]
- Extensive code comparison is required if bespoke codes are used in the design. [The principal codes and standards adopted are UK RGP. The applicability of these codes and standards on the UK context, to identify significant gaps on the RGP application of these codes and standards will be considered]

- Where no appropriate established codes or standards are available extensive justification of the use of similar codes and demonstration of the reliability achieved by their use is required. [The SMR-300 adopts appropriate and established codes and standards]
- The RP should consider that ONR has different assumptions regarding the type of aircraft involved in the impact load case. [Will be considered in response to relevant external hazards SAPs]
- The RP should consider the cliff edge effects from combined hazards. [Will be considered in response to relevant external hazards SAPs]
- There are different assumptions on incident loads informed by internal hazards. [Will be considered in response to relevant internal hazards SAPs]
- ONR requires the RP to understand (and mitigate) the risks associated with construction, commissioning, operations and decommissioning of the plant. [Will be considered in response to relevant engineering principles and decommissioning SAPs]
- ONR requires a demonstration that the risks of failure of the concrete containment are ALARP and the design has sufficient margin. RP has to confirm the margins on the concrete containment (ultimate pressure capacity). [The SMR-300 employs a metal containment structure protected by an SC containment enclosure structure. Will be considered in response to relevant integrity of metal components and structures, and civil engineering SAPs]
- Modern approaches to the design for fire safety of novel forms of construction are required. [Will be considered in response to relevant containment and ventilation, and civil engineering SAPs]

The lessons learnt which are summarised above are being considered and will be responded to in Revision 1 of this chapter. The design principles and current development of the Generic SMR-300 provide confidence that these lessons, where relevant, can be addressed successfully.

20.4.3 CAE Summary

The Generic SMR-300 Civil Engineering design and analyses has been undertaken using best practice nuclear industry codes and standards and RGP from recent UK GDA submissions. These codes and standards include guidance and expectations from the US NRC, ONR and design and analysis codes from US professional engineering organisations (ACI, ANSI/AISC, ASCE/SEI and ASME).

20.5 DESIGN OF CIVIL SSCs

Claim 2.2.11.2: Civil SSCs are designed using best practice analysis methodologies, taking cognisance of RGP and OPEX.

This subchapter outlines the analysis methodologies used in the design of SMR-300 Civil Structures. It considers:

- **Safety Functions of Civil Structures**, which determine the classification of Civil Structures;
- **Classification of Civil Structures and SSCs**, which determines the appropriate design codes and standards;
- **Design Codes and Standards** applied to the Civil Engineering SSCs;
- **Analysis and Design Methodology** to demonstrate that the design meets the requirements of the design codes and standards.

20.5.1 Safety Functions of Civil Structures

PSR Part A Chapter 2 General Design Aspects and Site Characteristics [4] introduces the SMR-300 high-level plant functions. The Civil Engineering SSCs with high-level plant functions are presented in Table 5.

Table 5: High-Level Plant Functions for Civil Engineering SSCs

SSC Name	SSC Code	SMR Class	High-Level Function
Containment Structure	CS	B	• Containment Integrity • Environmental Protection
Containment Enclosure Structure	CES	C	• Containment Integrity • Environmental Protection
Annular Reservoir	AR	C	• Post-Accident Heat Removal
Reactor Auxiliary Building Structure	RAB	C	• Containment Integrity • Environmental Protection
Intermediate Building	IB	C (segment adjacent to the CES) D (segment adjacent to the Turbine Building)	• Containment Integrity • Environmental Protection
Radioactive Waste Building Structure	RWB	D	• Containment Integrity • Environmental Protection

The safety classification of a function is either safety-related or non-safety related based upon the design function it performs during a Design Basis Event (DBE).

A safety function is defined as a function that is relied upon during or following a DBE to ensure:

- The integrity of the Reactor Coolant Pressure Boundary (RCBP).
- The capability to shut down the reactor and maintain it in a safe shutdown condition; or

- The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guideline exposure of Title 10 CFR Part 50.34(a)(1) [38], Title 10 CFR Part 50.67(b)(2) [39], or Title 10 CFR Part 100.11 [40], as applicable.

All other functions are classed as non-safety functions. The safety and non-safety functions of each Civil Engineering SSC are presented in Appendix B.

20.5.2 SMR Class and Seismic Category of Civil Structures and SSCs

The SMR-300 design approach is based on meeting the applicable sections of Title 10 CFR Part 50 Appendix A General Design Criteria [19] relating to classification by complying with the applicable requirements of the US NRC Regulatory Guides and U.S. Nuclear Regulatory Commission Technical Report Designation (NUREG)-0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Plants [41]. The classification system provides a means of identifying the extent to which SSCs provide safety-related and seismic functions and provides a way to designate applicable codes and standards to SSCs.

SSCs are classified as either safety-related or non-safety-related based on the design function they perform during a DBE. Safety-related and non-safety-related SSCs may be subject to other classifications based on the function they perform or support. For example, mechanical SSCs may have a pressure integrity classification and a seismic classification. The Holtec SMR classification system uses classes from A to F depending on whether the SSC is safety related and what its quality classification is. The SMR Class system is detailed in PSR Part A Chapter 2 General Design Aspects and Site Characteristics [4].

Furthermore, GDC 2 requires that nuclear power plant "*Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions*". Title 10 CFR Part 50 Appendix S Earthquake Engineering Criteria for Nuclear Power Plants [20] contains the criteria to which the plant design bases demonstrate the capability to function during and after vibratory ground motion associated with the Safe Shutdown Earthquake (SSE) conditions.

The seismic classification methodology used within the SMR Class system complies with the preceding criteria, as well as with recommendations stated within Regulatory Guide 1.29 Seismic Design Classification for Nuclear Power Plants [23]. The methodology classifies structures, systems, and components into three categories:

- Seismic category I (C-I)
- Seismic category II (C-II)
- Non-seismic (NS)

Generally, seismic category C-I applies to both functionality and integrity, whereas seismic category C-II applies only to integrity. Seismic category C-I applies to safety-related SSCs and those required to support or protect safety-related SSCs. Category C-II applies to SSCs that need to be designed to preclude their structural failure during SSE conditions or interaction with C-I SSCs.

Note that, the C-I designation does not imply that an SSC is safety-related. It is a specific classification used to identify the SSCs that are designed to withstand the effects of the SSE. However, all safety-related items are C-I.

Table 6 provides the classification of the Civil Engineering SSCs within the scope of this PSR chapter and their associated seismic category.

Table 6: SMR Class and Seismic Categories for Civil Engineering SSCs

Structure	SMR Class	Safety Classification	Seismic Category
Containment Structure	B	Safety-related	C-I
Containment Enclosure Structure	C	Safety-related	C-I
Annular Reservoir	C	Safety-related	C-I
Reactor Auxiliary Building	C	Safety-related	C-I
Intermediate Building	C (segment adjacent to the CES) D (segment adjacent to the Turbine Building)	Safety-related Non-safety-related	C-I NS
Radioactive Waste Building	D	Non-safety-related	NS*

* RWB is designed to withstand earthquake loading other than SSE in accordance with US NRC RG 1.143 [17].

Note that Holtec acknowledge the differences in the approach to safety categorisation and classification between the US NRC requirements and other national and international standards. This is discussed in PSR Part A Chapter 2 and will be developed during PSR Revision 1 to address gaps identified in the SSEC Revision 0 relating to categorisation and classification to ensure regulatory expectations are met.

20.5.3 Design Codes and Standards Applied to Civil SSCs

Subchapter 20.4 sets out the design codes and standards used for the SMR-300 Standard Design and identifies the applicable UK RGP for the Generic SMR-300.

It is important to reiterate that no novel or in-house (bespoke) design codes have been applied in the development and design of the SMR-300. The principal codes and standards applied to the design of nuclear safety related Civil Engineering SSCs are nuclear-specific and are considered RGP by the UK nuclear industry.

The principal Civil Engineering design codes and standards for SSCs within the scope of this chapter are summarised in Table 7.

Table 7: Principal Codes and Standards applied to Civil Engineering SSCs

SSC/Technical topic	SMR-300 Codes and Standards
Containment vessel	ASME BPVC.III.1.NE [32] ASME BPVC.III.NCA [33]

Steel-plate-Composite (SC) structures	ANSI/AISC N690-18, Appendix N9 [30]
Reinforced concrete structures	ACI 349-13 [29]
Steel structures	ANSI/AISC N690-18 [30]
Seismic analysis	ASCE/SEI 4-16 [31]

20.5.4 Analysis and Design Methodology

Civil structures are designed and analysed for normal operation loads and design basis loads arising from internal hazards, external hazards, and internal plant faults. The materials, loads, load combinations and design methodologies considered for the SMR-300 design are described in the following sub-chapters.

The structural acceptance criteria will be in accordance with the requirements of the ASME BPVC, Section III, Division 1, Subsection NE Class MC Components [32].

20.5.4.1 Materials

The materials used for the design and construction of Civil Engineering SSCs are specified according to the requirements of the applicable design codes and standards.

(REDACTED)

20.5.4.2 Loads and Load Combinations

The loads applied for the design of the Civil Engineering structures are derived from normal operation, internal hazards, external hazards and test loads.

Normal operational loads are the maximum loads induced during normal plant operation and shutdown. The external hazards considered in the generic SMR-300 design are derived in PSR Part B Chapter 21 External Hazards [43]. The internal hazards considered in the generic SMR-300 design are derived in PSR Part B Chapter 22 Internal Hazards [44].

Load combinations are consistent with the requirements of the applicable codes and standards and US NRC requirements.

20.5.4.2.1 Containment Structure

Table 8 presents the loads applied to the structural analysis and design of the CS. Loads and load combinations are in accordance with RG 1.57 Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components [24] and NUREG-0800, Standard Review Plan Section 3.8.2 Steel Containment [27].

Table 8: Containment Structure Loads

Load	Description
D	Dead loads
L	Live loads including those transferred from platform and crane

Load	Description
P_t	Test pressure
T_t	Test temperature
T_o	Governing thermal loads during startup, normal operating, or shutdown condition
R_o	Governing pipe reaction loads during startup, normal operating, or shutdown conditions
P_o	External pressure loads due to pressure vibration inside or outside containment
E	Loads generated by the Operating Basis Earthquake (OBE) plus sloshing effects
E’	Loads generated by the SSE plus sloshing effects
P_a	Pressure loads due to pipe break, pool swell, and subsequent hydrodynamic loads
T_a	Thermal loads due to pipe break, pool swell, and subsequent hydrodynamic loads
R_a	Pipe reaction due to pipe break, pool swell, and subsequent hydrodynamic loads
P_s	All pressure loads due to Steam Relief Valve (SRV) discharge plus pool swell, and hydrodynamic loads
T_s	All thermal loads due to SRV discharge plus pool swell, and hydrodynamic thermal loads
R_s	All pipe reaction loads due to SRV discharge plus pool swell, and hydrodynamic reaction loads
Y_r	Equivalent static load from reaction on broken pipe during DBA
Y_j	Jet impingement equivalent static load from broken pipe during the DBA
Y_m	Missile impact equivalent static load due to or during DBA (e.g., pipe whipping)
F_L	Load due to post LOCA flooding of the containment
Pg_1	Pressure due to 100% fuel clad metal/water reaction
Pg_2	Pressure resulting from uncontrolled hydrogen burning
Pg_3	Pressure due to post-accident carbon dioxide (assumed agent) inerting

20.5.4.2.2 Seismic Category I Concrete Structures

Table 9 presents the loads applied to the structural analysis and design of Seismic Category I concrete structures. Loads and load combinations are based on ACI 349 [29] as modified by RG 1.142 Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containment) [22].

Table 9: Seismic Category I Concrete Structures Loads

Load	Description
D	Dead loads, or related internal moments and forces, including piping, and equipment dead loads
F	Loads due to weight and pressures of fluids, or related internal moments and forces
H	Loads due to weight and pressure of soil, water in soil, or other materials

Load	Description
L	Live loads due to occupancy and moveable equipment
R_o	Piping and equipment reaction loads, excluding dead load and earthquake reactions
R_a	Piping and equipment reaction loads under thermal conditions generated by a postulated pipe break and including Ro
T_o	Thermal loads caused by temperature distributions within the concrete structure as a result of normal operating or shutdown conditions
T_a	Thermal loads caused by temperature distributions within the concrete structure as a result of accident conditions generated by a postulated pipe break and including T_o
W	Operating basis severe wind loads
W_t	Extreme wind loads due to the design basis tornado and hurricane loads, including velocity pressure, tornado wind pressure, tornado created differential pressure, and tornado/hurricane generated missiles
E_o	Seismic load due to an OBE including OBE-induced piping and equipment reactions
E_{ss}	Seismic load due to a SSE including SSE-induced piping and equipment reactions
C_{cr}	Loads due to building crane
P_a	Pressure loads due to accident conditions generated by a postulated pipe break
Y_j	Jet impingement load generated by a postulated pipe break
Y_r	Loads generated by the reaction of the broken pipe during a postulated break
Y_m	Missile impact loads generated by a postulated pike break, such as pipe whip
F_b	Buoyant force of the design basis flood

20.5.4.2.3 Seismic Category I Steel Structures

In addition to the CS, SMR-300 design Seismic Category I steel structures also include the polar crane support structure, the steam generator and pressurizer support frame, and some miscellaneous structures.

Table 10 presents the loads applied to the structural analysis and design of Seismic Category I steel structures. Loads and load combinations are based on ANSI/AISC N690-18 [30] and in accordance with NUREG-0800, Standard Review Plan Section 3.8.4 Other Seismic Category I Structures [28].

Table 10: Seismic Category I (other than the CS) Steel Structures Loads

Load	Description
D	Dead loads due to the weight of the structural elements, fixed-position equipment, and other permanent appurtenant items; weight of crane trolley and bridge
C	Rated capacity of crane, including the maximum wheel loads of the crane and the vertical, lateral and longitudinal forces induced by the moving crane

Load	Description
F	Loads due to weight and pressures of fluids with well-defined densities and controllable maximum height
H	Loads due to weight and pressure of soil, water in soil, or bulk materials
L	Live loads due to occupancy and moveable equipment, including impact
L_r	Roof live load
R	Rain load
R_o	Pipe reactions during normal operation, start-up or shutdown conditions, based on the most critical transient steady-state condition
R_a	Pipe and equipment reactions generated by the postulated accident, including R_o
S	Snow load as stipulated in ASCE/SEI 7 Minimum Design Loads in Buildings and Other Structures [45] for Risk Category IV facilities
T_o	Thermal loads caused by temperature distributions within the concrete structure as a result of normal operating or shutdown conditions
T_a	Thermal loads generated by the postulated accident, including T_o
W	Wind load as specified in ASCE/SEI 7 [45] for Risk Category IV facilities, or as specified by the Authority Having Jurisdiction (AHJ)
W_t	Loads generated by the specified design (basis) tornado, including wind pressure, pressure differentials, and tornado-borne missiles, as defined in NUREG-0800, SRP Section 3.3.2 Tornado Loadings [26] or as specified by the AHJ
E_o	Seismic load due to an OBE including OBE-induced piping and equipment reactions
E_s	Seismic load due to a SSE including SSE-induced piping and equipment reactions
P_a	Maximum differential pressure load generated by the postulated accident
Y_j	Jet impingement load generated by the postulated accident
Y_r	Loads on the structure generated by the reaction of the broken high-energy pipe during the postulated accident
Y_m	Missile impact loads generated by a postulated pike break, such as pipe whip

20.5.4.2.4 Seismic Category I Steel-Plate Composite Structures

The Seismic Category I CES of the generic SMR-300 design is required to provide adequate radiation shielding and resistance to severe external loads such as tornado missile and crashing aircraft impacts.

Table 11 presents the loads applied to the analysis and design of the CES. Loads and load combinations have been derived from Chapter NB2 of ANSI/AISC N690-18 [30].

Table 11: Seismic Category I Steel-Plate Composite Structure Loads

Load	Description
D	Dead loads due to the weight of the structural elements, fixed-position equipment, and other permanent appurtenant items; weight of crane trolley and bridge
C	Rated capacity of crane, including the maximum wheel loads of the crane and the vertical, lateral and longitudinal forces induced by the moving crane
F	Loads due to weight and pressures of fluids with well-defined densities and controllable maximum height
H	Loads due to weight and pressure of soil, water in soil, or bulk materials
L	Live loads due to occupancy and moveable equipment, including impact

Load	Description
L_r	Roof live load
R	Rain load
R_o	Pipe reactions during normal operation, start-up or shutdown conditions, based on the most critical transient steady-state condition
R_a	Pipe and equipment reactions generated by the postulated accident, including R_o
S	Snow load as stipulated in ASCE/SEI 7 [45] for Risk Category IV facilities
T_o	Thermal loads caused by temperature distributions within the concrete structure as a result of normal operating or shutdown conditions
T_a	Thermal loads generated by the postulated accident, including T_o
W	Wind load as specified in ASCE/SEI 7 [45] for Risk Category IV facilities, or as specified by the AHJ
W_t	Loads generated by the specified design (basis) tornado, including wind pressure, pressure differentials, and tornado-borne missiles, as defined in NUREG-0800, SRP Section 3.3.2 Tornado Loadings [26] or as specified by the AHJ
E_o	Seismic load due to an OBE including OBE-induced piping and equipment reactions
E_s	Seismic load due to a SSE including SSE-induced piping and equipment reactions
P_a	Maximum differential pressure load generated by the postulated accident
Y_j	Jet impingement load generated by the postulated accident
Y_r	Loads on the structure generated by the reaction of the broken high-energy pipe during the postulated accident
Y_m	Missile impact loads generated by a postulated pike break, such as pipe whip

20.5.4.3 Structural modelling

Civil Engineering structures are analysed using three-dimensional finite element modelling techniques for both static and dynamic analyses. The mass of structures includes self-weight, the mass of equipment, distribution systems, effective live loads, etc. The stiffness of the structural components is a function of the geometric and material properties of the component.

The types of finite element analyses implemented are determined by the type of the structure and the response parameters of interest.

20.5.4.4 Seismic Analysis

The characterization of the SMR design basis earthquake utilises seismic design response spectra (SDRS), which consist of a modified RG 1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants [25], anchored at a peak ground acceleration (PGA) of 0.40g in each orthogonal direction. Notably, the SDRS are anchored at the base of the CES at an elevation of almost 90 feet below grade.

Refer to the Generic Site Envelope Report [46] for commentary comparing the SMR-300 SDRS against the spectra derived to define the SMR-300 Great Britain Generic Site Envelope (GB GSE).

A time-domain nonlinear Soil-Structure Interaction (SSI) analysis methodology has been employed to assess the seismic responses of the seismic Category I structures and the neighbouring structures of the SMR-300.

The outcomes of the SSI analysis have direct implications for subsequent structural evaluations, encompassing the qualification of designs for these structures, as well as defining seismic loads for assessing safety-related equipment and their supporting structures within the said edifices.

Holtec's SSI analysis method is adept at explicitly capturing geometric nonlinearity at the interfaces connecting the deeply embedded seismic Category I structures with the surrounding soil. Additionally, it can effectively model the pronounced soil nonlinearity that is anticipated to manifest in close proximity to these embedded structures. Notably, both these nonlinear effects can have substantial significance in scenarios involving strong seismic activity and/or weakened soil conditions.

The analysis is conducted using the explicit finite element software, LS-DYNA. The LS-DYNA SSI model is composed of several key components, encompassing the soil, seismic structures, and adjacent non-seismic structures.

The LS-DYNA SSI model is characterized by the following attributes:

- Solid elements are used to model soil and thick civil structure components.
- Water in the annular reservoir and in the spent fuel pool is modelled using solid elements with a simple fluid material model which has no shear capacity.
- Shell or thick shell elements are used to model walls and slabs of civil structures or thin steel equipment/structures.
- Beam elements are used to model beams and columns of civil or steel structures and certain equipment.
- Mass elements are used to account for the masses of equipment of significant weight.
- The model explicitly captures the soil nonlinear behaviour and the geometric nonlinearity at the soil/structure contact interface.
- Structures are modelled using linear elasticity.

The dynamic finite element models are developed and validated using a staged approach.

20.5.4.5 Aircraft Impact Assessment

The SMR-300 Generic Design considers aircraft impact according to the guidance adopted from the US NRC. For any new reactor designed after July 13, 2009, Title 10 CFR Part 50.150 [47] requires a design-specific assessment of the effects on the facility from the beyond-design-basis impact of a large, commercial aircraft used for long distance flights in the United States.

The following general requirements apply for the generic SMR-300 design following aircraft impact:

a) The reactor core shall remain coolable.
b) The containment shall remain intact.
c) SFP cooling and SFP integrity is maintained.

d) The following structures shall be demonstrated that they are able to continue to perform their safety functions following the external event:
    a. CES and CS.
    b. MCR and RSF areas of the RAB.
e) Missile effects on important to safety SSC shall be considered.
f) Safety functions of safety systems shall not be degraded, such as due to the effects of the external event on nearby SSC or consequential missile hazards.
g) Support systems integrity shall be maintained such that there is no degradation of the availability of fundamental safety functions following the external event.

Further information on this topic is provided in PSR Part B Chapter 21 External Hazards [43], where a discussion on malevolent and non-malevolent aircraft impacts is provided.

20.5.4.6 Foundation Design

Loads transferred to foundation structures through the structure and soil and explicitly accounted for and represented via the 3D modelling of the SMR-300.

Loads and load combinations applied to foundation structures are in accordance with the applicable codes and standards of the SMR-300.

The design of the foundation structures is undertaken in compliance with the codes and standards indicated in Table 7.

20.5.5 CAE Summary

The SMR-300 civils design has been undertaken using best practice nuclear industry codes and standards by use of the ASME BPVC, ANSI/AISC, ACI, ASCE/SEI design codes.

The analysis methods used are also considered best practice, with the use of structural finite element analysis codes, including LS-DYNA for seismic analysis. These codes have been subject to appropriate verification and validation.

For normal load conditions and DBEs, all structures are shown to meet the appropriate structural acceptance criteria defined in the relevant design codes. For BDB Events, it is demonstrated that suitable margins exist and that there are no cliff-edge effects.

20.6 DEFENCE IN DEPTH

Claim 2.2.11.3: Defence in depth of Civil SSCs is maintained so far as is reasonably practicable through good engineering practice.

The principle of Defence in Depth requires that the design and analysis of a nuclear facility is based upon multiple independent levels where the failure of one of those levels is accommodated by the engineered safety features and safety margins within the design.

These levels are defined in the ONR Safety Assessment Principles (SAPs) for Nuclear Facilities [35] as follows:

- **Level 1 Prevention of abnormal operation and failures by design**: Conservative design, construction, maintenance and operation in accordance with appropriate safety margins, engineering practices and quality levels.
- **Level 2 Prevention and control of abnormal operation and detection of failures**: Control, indication, alarm systems or other systems and operating procedures to prevent or minimise damage from failures.
- **Level 3 Control of faults within the design basis to protect against escalation to an accident**: Engineered safety features, multiple barriers and accident or fault control procedures.
- **Level 4 Control of severe plant conditions in which the design basis may be exceeded, including protecting against further fault escalation and mitigation of the consequences of severe accidents**: Additional measures and procedures to protect against or mitigate fault progression and for accident management.
- **Level 5 Mitigation of radiological consequences of significant releases of radioactive material**: Emergency control and on- and off-site emergency response.

The following describes how Defence in Depth is demonstrated for the Civil Engineering topic.

20.6.1 Conservative Design

The civils design of the SMR-300 reactor is undertaken using analysis methods that are considered RGP, using conservative material properties and load cases (including load combinations) that meet the requirements of ASME BPVC. The structural acceptance criteria will be defined within PSR Revision 1.

20.6.2 Design Basis Analysis

For faults within the design basis, the civil engineering structures will be analysed to demonstrate that the design is robust during all design basis load cases, including natural and human-induced internal and external hazards. The load cases are defined in subchapter 20.5.4.2 and the basis of the structural modelling described in subchapter 20.5.4.3.

20.6.3 Beyond Design Basis Accident Analysis

The Beyond Design Basis Accident (BDBA) analysis in PSR Part B Chapter 15 Beyond Design Basis, Severe Accident Analysis and Emergency Preparedness [13] considers the design response to a hazard whose magnitude is slightly above the design basis and demonstrates an acceptable margin to failure. The BDBA analysis will consider both internal reactor hazards and natural and human-induced internal and external hazards. The BDBA analysis will quantify those margins in terms of hazard magnitude or SSC withstand.

20.6.4 CAE Summary

The key requirement of the Civil SSCs is to protect the reactor and its support systems from internal and external hazards, both natural and human-induced, and to provide containment and shielding to protect people and the environment.

The approach to internal and external hazards is to design the reactor structures against Design Basis and BDB events caused by natural and human induced events. This ensures that the civil design is robust to all natural and human induced hazards within the Design Basis and is robust against cliff-edge effects due to BDB events.

The physical design and layout of the structures includes features to prevent non-ductile failure of the containment structure during seismic events, building levels and design features to prevent water ingress into the buildings and robust structures to prevent failure due to wind and snow loadings and missiles.

20.7 QUALITY MANUFACTURING AND INSTALLATION

Claim 2.2.11.4: Civil SSCs achieve the design intent through quality manufacturing and installation process.

This subchapter considers:

- Quality Assurance
- Construction/Fabrication
- EIMT

20.7.1 Quality Assurance

The quality assurance requirements for civil structures can be found in the Design Specifications for the respective buildings and structures. These include requirements for:

- Materials and Coatings
- Construction and Fabrication
- Pre-operational structural proof tests
- Containment Leak Tests

The requirements are generally taken from the applicable ASME BPVC as defined in Table 7.

A complete list of Design Specifications for the SMR-300 will be provided at PSR Revision 1.

20.7.2 Construction/Fabrication

The Generic SMR-300 design utilises modular construction techniques where possible for the twin unit site. The construction of the CS, CES and some areas of the RAB are standardised to the extent possible.

The CES is to be of SC construction instead of RC, which has been traditionally used in the construction of nuclear containment enclosure structures. The SC form of construction provides the following benefits over traditional reinforced concrete:

- Eliminates the need for form work and rebar assembly on site.
- Single code of construction tolerance since modules are fabricated in the shop.
- Reduction of construction overheads, since the modules are fabricated in the shop.

The overall outcome of the above benefits is a reduced construction programme and greater quality control.

The basemat foundation of the CS and CES consists of a large RC raft foundation containing a significant quantity of steel reinforcement bars and embedded pipes/plates. A 3-D simulation model is used to identify and eliminate potential clashes between the steel reinforcement and embedded items. Temperature and crack control will be important considerations in the design, detailing and construction of the basemat foundation.

Further information on the SMR-300 construction approach is provided in PSR Part B Chapter 25 Construction and Commissioning Approach [9].

20.7.3 Examination, Inspection, Maintenance and Testing (EIMT)

The operational design life of the Generic SMR-300 design is 80 years. However, the Civil Engineering structures will need to perform their respective safety functions beyond the 80-year operational design life to account for the construction and decommissioning phases.

An effective EIMT programme shall be in place throughout the lifetime of the facility. Further information on the EIMT programme is provided in PSR Part B Chapter 9 Description of Operational Aspects and Conduct of Operations [8].

Pre-operational structural proof tests and containment leak tests will be undertaken during SMR-300 construction.

20.7.4 CAE Summary

The quality assurance requirements for civil structures are defined in the Design Specifications for the respective buildings and structures.

Construction and fabrication techniques will ensure that any defects during build are minimised to reduce the need for corrective action. This will be ensured through build quality assurance documentation.

EIMT will demonstrate the fitness for purpose of civil structures by pre-operational structural proof tests and containment leak tests and by through-life inspections. This will include the non-destructive examination of civil structures.

20.8 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the Civil Engineering Chapter and how this Chapter contributes to the overall demonstration of ALARP for the generic SMR-300. PSR Part A Chapter 5 Summary of ALARP [15] sets out the overall approach for demonstration of ALARP and how contributions from individual chapters are consolidated.

This subchapter therefore consists of the following elements:

- Technical Summary
- ALARP Summary
  - Review against Relevant RGP
  - Demonstration Against Risk Targets
    - Evaluation of Risk
  - Risk Reduction Options
  - GDA Commitments and Forward Actions
- Conclusion

A review against these elements is presented below under the corresponding headings.

20.8.1 Technical Summary

PSR Chapter B Part 20 states that the Civil Engineering SSCs within the scope of this report will meet the high-level Claims of the SSEC and that the SSCs can be substantiated at Pre-Construction Safety Report (PCSR) stage. This is demonstrated through the following sub-claims:

Claim 2.2.11: The overall design and architecture of Civil structures ensure that safety functions and non-safety functions are delivered and faults arising from failures of structures are minimised.

The SMR-300 design provides multiple physical barriers to the release of radioactivity to the environment. These include the fuel cladding, the reactor coolant pressure boundary, and the containment structure. It is the intent that during normal operations and anticipated operational occurrences the SMR-300 design prevents challenges to the integrity of all these barriers and the design incorporates highly reliable passive engineered safety features (along with an Instrumentation and Control (I&C) system with multiple levels of anticipatory reactor trip signals).

The containment of the SMR-300 is designed to remain intact and sealed during all postulated events, and to reject its internal energy to the large body of water surrounding the containment structure. After a postulated event, such as a LOCA, the plant is designed to automatically achieve and maintain a safe shutdown state without operator actions, without external water, without external power, and without active systems.

The safety basis incorporates defence-in-depth via multiple and diverse simple pathways for heat rejection from the core. All safety systems are located within the CES, making them secure and safe from postulated external events and threats.

The SMR-300 civils design has been undertaken using best practice nuclear industry codes and standards by use of the ASME BPVC, ANSI/AISC, ACI, ASCE/SEI design codes.

The analysis methods used are also considered best practice, with the use of structural finite element analysis codes, including LS-DYNA for seismic analysis. These codes have been subject to appropriate verification and validation.

For normal load conditions and DBEs, all structures are designed to meet the appropriate structural acceptance criteria defined in the relevant design codes. For BDB Events, it will be demonstrated that suitable margins exist and that there are no cliff-edge effects.

The quality assurance requirements for civil structures are defined in the Design Specifications for the respective buildings and structures.

Construction and fabrication techniques will ensure that any defects during build are minimised to reduce the need for corrective action. This will be ensured through build quality assurance documentation.

EIMT will demonstrate the fitness for purpose of civil structures by pre-operational structural proof tests and containment leak tests and by through life inspections. This will include the non-destructive examination of civil structures.

The key requirement of the Civil SSCs is to protect the reactor and its support systems from internal and external hazards, both natural and human induced. The approach to internal and external hazards is to design the reactor structures against Design Basis and BDB events caused by natural and human induced events. This ensures that the civil design is robust to all natural and human induced hazards within the Design Basis and is robust against cliff-edge effects due to BDB events. The physical design and layout of the structures includes features to prevent non-ductile failure of the containment structure during seismic events, building levels and design features to prevent water ingress into the buildings and robust structures to prevent failure due to wind and snow loadings and missiles.

PSR Revision 1 will define the hazards considered in the design and their magnitude by reference to PSR Part B Chapter 21 External Hazards [43] and PSR Part B Chapter 22 Internal Hazards [44]. It will present a summary of the structural analysis undertaken for the SMR-300 Reference Design and will identify any gaps.

20.8.2 ALARP Summary

20.8.2.1 Demonstration of RGP

The design of the SMR-300 civil structures complies with RGP and US NRC requirements applicable in the US. The design adopts nuclear-specific codes and standards endorsed by the US NRC and internationally recognised bodies such as International Atomic Energy Agency (IAEA). The principal codes and standards identified within subchapter 20.4 are considered RGP by the UK nuclear industry. This is based on existing practices adopted on UK nuclear licensed sites, application in earlier and successful GDAs, as well as recognition as RGP by ONR TAGs.

A significant portion of the reactor building is partially embedded below grade. The design methodologies will adopt tried-and-tested civil engineering approaches for the construction of the reactor building with focus on the below grade external walls and basemat. The design of the CES adopts Steel-Concrete Composite (SC) modular construction and the design basis and testing are in development. Novel methodologies will be avoided, where possible, with adherence to RGP. Furthermore, the AR is formed by the inner and outer walls of the CES and CS respectively, which are designed to different codes and standards. To ensure that the AR remains leak-tight, appropriate codes and standards will be deployed.

Forward actions will form the basis for setting out the process to justify any gaps from UK RGP. Forward Actions have been collated and are managed via the process described in PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [48].

20.8.2.2 Demonstration Against Risk Targets

The numerical targets against which the demonstration ALARP is considered can be found in PSR Part A Chapter 2 [4]. Civil SSCs, through the defined safety functions, will contribute to the demonstration of ALARP by comparison against the risk targets in two ways:

- By fulfilling safety functions for normal operations (e.g., shielding and containment), and thereby contributing to achieving Targets 1-3.
- By achieving their safety classification as a duty system or a protection system, where claimed, they will contribute to the achievement of accident risk, Targets 4-9.

Risks below the Basic Safety Objectives (BSOs) are considered broadly acceptable, and no further consideration of risk reduction is required. Risks between the BSOs and Basic Safety Levels (BSLs) require a consideration of risk reduction options. Risks above the BSLs are not acceptable.

20.8.2.2.1 Evaluation of Risk

Evaluation of nuclear safety risk is not applicable to Civil Engineering SSCs. The safety classification of the Civil Engineering SSCs will be associated with an annual probability of failure or probability of failure on demand (PFD), which is then used to calculate the overall comparison against the nuclear safety risk targets as described above. Depending on the type of construction and the importance of the structure, the failure probability used in the calculations will be different.

At this time, the evaluation of the normal operations and accident risks against Targets 1-9 has not been provided. This information will be presented in PSR Part B Chapter 10 Radiological Protection [11] for normal operations, and PSR Part B Chapter 14 Design Basis Accident Analysis [12], PSR Part B Chapter 15 Beyond Design Basis and Severe Accident Analysis, and Emergency Preparedness [13], and PSR Part B Chapter 16 Probabilistic Safety Analysis [14] for accident conditions.

20.8.2.3 Risk Reduction Options

This is a placeholder to identify and review relevant Position Papers and Design Decision Papers with a view to demonstrate which option(s) is/are ALARP.

It will summarise those option evaluations, and it will briefly explore if other risk reduction options have or could be considered and either:

- Present the ALARP argument for why those options have not been implemented.
- Present the ALARP argument for why those options will be implemented in future.
- Create a Forward Action to consider the option(s) at some future point (noting this still must be a point where a meaningful design improvement could be made).

The process for the assessment of risk reduction options is presented in the Generic Design Assessment Reference Design Process and GDA Prospective Design Change Register [49]. PSR Part A Chapter 5 ALARP Summary [15] considers the holistic risk-reduction process for the Generic SMR-300.

20.8.2.4 GDA Commitments and Forward Actions

There are no GDA commitments identified for PSR Part B Chapter 20.

Forward Actions have been collated and are managed via the process described in PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [48]. PSR Part A Chapter 5 ALARP Summary [15] describes the contribution of the forward actions to the ALARP argument.

20.8.3 Conclusion

The conclusion of this Chapter of the PSR is that:

- The Chapter Claims identified have been met to a maturity aligned with a preliminary safety report. Further claims, arguments and evidence will be presented in due course as the design develops.
- Safety and non-safety functions have been identified for the Civil SSCs.
- A systematic classification system has been applied to the SSCs commensurate with their importance.
- The classification system allows the appropriate design codes and standards to be identified.
- The Civil SSCs have been designed to meet UK RGP that being US codes and standards for the design of civil structures.
- The substantiation against the identified codes and standards is likely to result in a design that contributes to the demonstration that risks to people during normal operations and accident conditions are tolerable and ALARP.

PSR Part A Chapter 5 concludes that it can be demonstrated that the Generic SMR-300 reduces risks to ALARP and that the Fundamental Purpose of the SSEC has been fulfilled.

20.9 REFERENCES

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20.10 LIST OF APPENDICES

Appendix A CAE Route Map A-1

Appendix B High-Level, Safety and Non-Safety Functions for Civil Structures B-1

Appendix A CAE Route Map

A summary of the Safety, Security, and Environmental Case (SSEC) claims for the civil engineering area is presented in Table 12.

Table 12: PSR Part B Chapter 20 CAE Route Map

Overarching SSEC Claim	Chapter Claim/s	Chapter Sub-Claim/s	Subchapter
Claim 2.2 – System / Process Design and Substantiation The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.	**Claim 2.2.11** The overall design and architecture of Civil SSCs ensure that safety functions and non-safety functions are delivered and faults arising from failures of the SSCs are minimised.	**Claim 2.2.11.1** Civil SSCs are designed using appropriate Codes and Standards, taking cognisance of Relevant Good Practice (RGP) and Operational Experience (OPEX).	20.4 CODES, STANDARDS AND METHODOLOGIES
		Claim 2.2.11.2 Civil SSCs are designed using best practice analysis methodologies, taking cognisance of RGP and OPEX.	20.5 DESIGN OF CIVIL SSCs
		Claim 2.2.11.3 Defence in depth of Civil SSCs is maintained so far as is reasonably practicable through good engineering practice.	20.6 DEFENCE IN DEPTH
		Claim 2.2.11.4 Civil SSCs achieve the design intent through quality manufacturing and installation process.	20.7 QUALITY MANUFACTURING AND INSTALLATION

Appendix B High-Level, Safety and Non-Safety Functions for Civil Structures

SSC	High-Level Function	SSC Description	SMR Class	Safety Function/Non-Safety Function
CS	Containment Integrity Environmental Protection	Containment Structure	B	TBC
CES	Containment Integrity Environmental Protection	Containment Enclosure Structure	C	Safety functions: • Provide support to Seismic Category I systems, structures and components (SSCs), housed inside the CES. • Provide radiation shielding. • Act as physical barrier and provide protection to the CS and other SSCs enclosed within the CES against environmental hazards, such as tornado and heavy wind. Non-safety functions: • Provide support to the SSCs of the Passive Containment Heat Removal System (PCH). • Provide support for removal and replacement of steam generator, if necessary. • Provide access to the AR to inspect the external surface of the CS, components of Passive Decay Heat Removal System (PDHR) and Secondary Decay Heat Removal System (SDHR) submerged in the AR. • Provide support for containment hatches, doors, and airlocks for equipment and personnel access. • Provide access for containment penetrations (mechanical piping, electrical, etc.). • Provide appropriate venting system for the AR to achieve adequate cooling for indefinite period.
AR	Post-Accident Heat Removal	Annular Reservoir	C	The AR is a component of the PCHR and inherits the following PCHR safety functions: • Maintain containment pressure and temperature below its design limits. • Mitigate pressure-assisted release of fission products to the environment. • Preserve the CS integrity. • Maintain the CS required leak tightness.

SSC	High-Level Function	SSC Description	SMR Class	Safety Function/Non-Safety Function
RAB	Containment Integrity Environmental Protection	Reactor Auxiliary Building	C	Safety functions: • Provide protection and separation for the Seismic Category I structures, and mechanical and electrical equipment located outside the Containment Structure (CS): ○ Class 1E Battery Room and associated equipment ○ Containment Penetrations/Accessways ○ Safety-related Instrumentation and Control (I&C) Non-safety functions: • House systems, components and structures critical for the normal operation/shutdown of the plant such as RHR, CVCS makeup, spent fuel pool cooling, and radwaste systems. • Provide provisions to support spent fuel removal to interim dry storage. • Provide provisions to support new fuel receipt and storage. • Provide Heating, Ventilation, and Air Conditioning (HVAC) for all the components and structures of the RAB and containment purging. • Provide radiation protection for plant personnel from radioactive components and minimization of contamination and radioactive waste generation, by incorporating the following design features to ensure compliance with applicable regulations such as Title 10 CFR Part 20.1101 [50] and Title 10 CFR Part 20.1406 [51] correspondingly: ○ Radiation zoning and access control ○ Remotely operated Process Instrumentation and Controls ○ Isolation and decontamination of substantial radiation sources ○ Minimization of accumulation of radioactive materials in Resin and Sludge Treatment Systems etc. • Provide suitable environments for personnel and equipment during all phases of plant operation. • Provide adequate laydown space to facilitate inspection, testing, and maintenance of structures, systems, and components located within the RAB.
IB	Containment Integrity Environmental Protection	Intermediate Building	C (segment adjacent to the CES) D (segment adjacent to the Turbine Building)	TBC

SSC	High-Level Function	SSC Description	SMR Class	Safety Function/Non-Safety Function
RWB	Containment Integrity Environmental Protection	Radioactive Waste Building	D	TBC