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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, commissioned, operated, and decommissioned on a generic site in the United Kingdom (UK) to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment as defined in Part A Chapter 1 Introduction [1].

The Fundamental Purpose is achieved through the Fundamental Objective of the Preliminary Safety Report (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] .

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

20.1.1 Purpose and Scope

The Overarching SSEC Claims are presented in Part A Chapter 3 [2].

This chapter (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 Part A Chapter 3 [2], 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 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 sub-chapter 20.3.

The scope of this chapter covers the following Civil Engineering SSCs within the Nuclear Island (NI) (refer to the GDA Scope Report [3]), as set out in sub-chapter 20.2:

- The Containment Enclosure Structure (CES)
- The Containment Structure (CS)
- The Reactor Auxiliary Building (RAB)
- The Intermediate Building (IB)

It is worth noting that the Annular Reservoir (AR), a component of the Passive Containment Heat Removal System (PCH), is structurally contained by the CES and CS. Therefore, the AR is a significant interface to the CES and CS.

Sub-chapter 20.4 covers the codes and standards associated with the design of Civil Engineering SSCs.

Sub-chapter 20.5 covers the safety functions, classification and analysis and design methodologies of Civil Engineering SSCs.

Sub-chapter 20.6 covers the defence in depth of Civil Engineering SSCs.

Sub-chapter 20.7 covers the quality manufacturing, installation, Examination Inspection Maintenance and Testing (EIMT) of Civil Engineering SSCs.

Finally, sub-chapter 20.8 provides a technical summary of how the claims for this chapter are achieved, together with a summary of the key contributions from this chapter to the ALARP principle. Sub-chapter 20.8 also discusses any GDA commitments that have arisen.

Excluded from the scope of Part B Chapter 20 is the Interim Spent Fuel Storage Installation (ISFSI) as its design development is still at an early stage [3]. Furthermore, internal structures and equipment supports are excluded from the scope of Part B Chapter 20. Structures that are not in the scope of the GDA but contain equipment which perform safety functions will be addressed in future safety reports.

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 (SC) modular construction and the design basis and testing of the SMR-300 SC concept are in development; furthermore, the AR is structurally contained by the inner and outer walls of the CES and CS, respectively.

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

20.1.2 Assumptions

Assumptions which relate to GDA topics have been formally captured in the Commitments, Assumptions and Requirements (CAR) process [5]. Further details of this process are provided in Part A Chapter 4 [6].

There are no assumptions raised in relation to Part B Chapter 20.

20.1.3 Interfaces with other SSEC Chapters

The Civil Engineering chapter interfaces with the following PSR chapters.

Civil Engineering SSCs are designed for the general design aspects and site characteristics of the generic SMR-300. The general design aspects, safety function classification and site characteristics are reported in Part A Chapter 2 [4].

Part A Chapter 4 presents the CAE for the Lifecycle Management for Safety and Quality Assurance of the generic SMR-300, which also encompass the Civil Engineering topic [6].

Civil Engineering SSCs enclose the reactor pressure vessel and fuel, core and reactor supporting SSCs. Part B Chapter 1 [7], Part B Chapter 2 [8] and Part B Chapter 5 [9] provide a description of the SSCs contained within the Civil Engineering SSCs of the generic SMR-300. Part B Chapter 1 [7] in particular, provides a description of the PCH, which is the system that contains the AR as a component.

Part B Chapter 9 [10] provides the relevant EIMT arrangements for the plant and states the limits and conditions of the safety case.

Part B Chapter 6 [11] provides a description of the Electrical Engineering SSCs enclosed within Civil Engineering structures.

Part B Chapter 10 [12], Part B Chapter 14 [13], Part B Chapter 15 [14], Part B Chapter 16 [15] provide a comparison against risk targets defined in Part B Chapter 2[8].

Part B Chapter 11 [16] describes how the fundamental objective for environmental protection will be achieved in the development of the environmental case, describes the interfaces between the PSR and the Preliminary Environmental Report (PER), and summarises the environmental aspects of the generic SMR-300.

Part B Chapter 12 [17] covers Nuclear Site Health and Safety and Conventional Fire Safety. The management of Nuclear Site Health and Safety applies to the full lifecycle of all Civil Engineering SSCs. Conventional Fire Safety is considered when establishing the layout of Civil Engineering SSCs.

Part B Chapter 17 [18] presents the CAE for Human Factors of the generic SMR-300.

Part B Chapter 18 [19] presents the CAE for the approach to structural integrity of SSCs. Several candidate higher reliability/very high reliability SSCs are derived from interaction with the CS.

Part B Chapter 19 [20] provides the functional requirements and design methodologies for Mechanical Engineering SSCs.

Part B Chapter 21 [21] provides an overview of the approach undertaken for the generic SMR-300 against the external hazards identified and characterized in the Generic Site Envelope Report (GSER) [22], evaluation of each hazard and derivation of UK generic site parameters.

Part B Chapter 22 [23] provides an overview of the approach to internal hazards for the generic SMR-300, identifies the relevant internal hazards and the SSCs that support claims relating to Internal Hazards.

Part B Chapter 23 [24] provides a description of the AR chemistry regime. The AR is in contact with the external surface of the CS and the internal surface of the CES. Exclusion of the CS and CES surfaces from the chemical environment of the AR is achieved through selection of adequate coating.

Part B Chapter 25 [25] provides the construction and commissioning approach of the generic SMR-300.

Part B Chapter 26 [26] provides the decommissioning approach of the generic SMR-300.

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.

A plan layout of the NI structures is given in Figure 1 which is extracted from the SMR-300 Plot Plan [27].

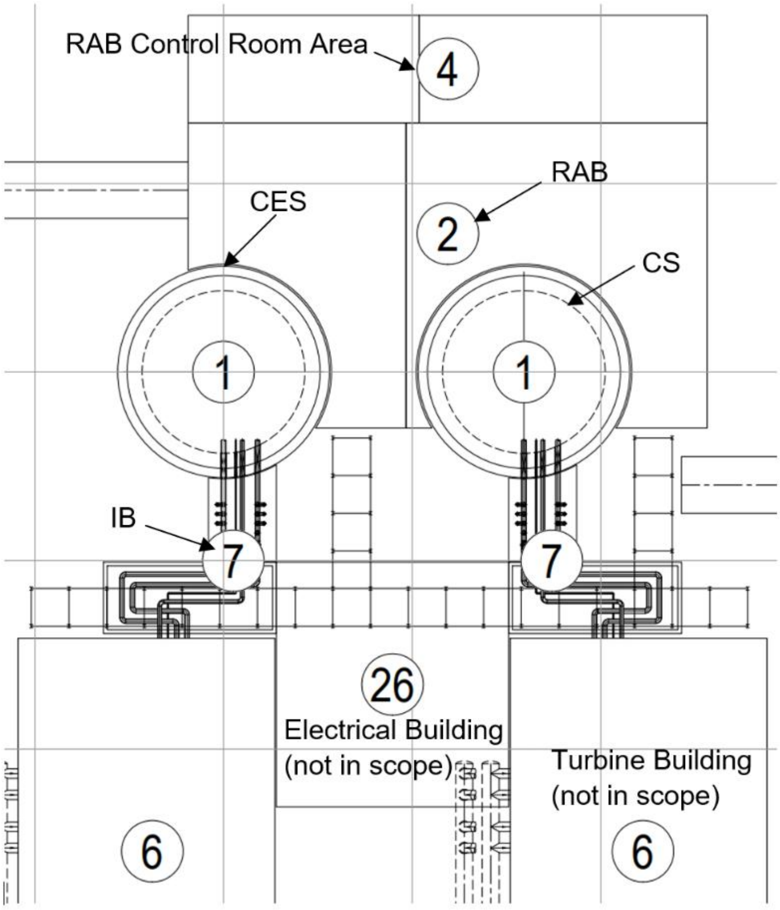

Figure 1: Plan Arrangement of Nuclear Island Structures [27]

Part A Chapter 2 [4] provides a description of the site layout and main buildings. The main Civil Engineering SSCs included within the NI are:

- The CES (number 1 on Figure 1) is described within sub-chapter 20.2.1.
- The CS (indicated by a dashed circle within the CES on Figure 1) is described within sub-chapter 20.2.2.
- The IB (number 7 on Figure 1) is described within sub-chapter 20.2.4.
- The RAB (number 2 and 4 on Figure 1) is described within sub-chapter 20.2.5.

The AR is not a Civil Engineering SSC, it is a component of the PCH. However, as the AR is structurally contained by the CES and CS and therefore a significant interface to Civil Engineering SSCs, a description of the AR is provided within sub-chapter 20.2.3.

As noted in sub-chapter 20.1, the ISFSI is excluded from the scope of this chapter and it is dealt within Part B Chapter 24 [28].

The classification of safety functions is described in Part A Chapter 2 [4] and the SMR-300 Structures, Systems and Components Classification Standard [29]. The classification methodology of the SMR-300 aligns and complies with United States Nuclear Regulatory Commission (USNRC) requirements. Holtec acknowledge the differences in the approach to safety categorisation and classification between USNRC requirements and other national and international standards. This is addressed in Part A Chapter 2 [4]. Holtec have developed an approach for categorisation and classification to meet UK Relevant Good Practice (RGP). This is defined in HI-2250210 Safety Assessment Handbook (SAH) [30]. The approach defined in the SAH will be applied post-GDA for the Civil Engineering SSCs within the scope of this chapter.

20.2.1 Containment Enclosure Structure

The CES is a cylindrical protective structure that fully envelops the CS (Figure 2). The CES is designed to protect the CS from external hazards, to provide radiation shielding around the CS, and to provide a water-tight enclosure for the AR.

The CES is a large deeply founded structure with its basemat at elevation [REDACTED] extending up to elevation [REDACTED] above grade.

The CES is designed to perform the following safety functions [31]:

- Provide support to Seismic Category I SSCs housed inside the CES.
- Provide shielding to plant and personnel from radioactive sources inside the CS during power operations and postulated accidents.
- Act as a physical barrier and provide protection to the CS and other SSCs enclosed within the CES from environmental hazards, such as tornado and heavy wind, and manmade hazards such as aircraft and missile impacts.

The CES is designed to perform the following non-safety functions [31]:

- Provide support to the SSCs of the PCH.
- Allow access for removal and replacement of the steam generator, if necessary.
- Allow access to the AR for inspection of the external surface of the CS and components of the Secondary Decay Heat Removal System (SDH) submerged in the AR.
- Provide support for containment hatches, doors and airlocks for equipment and personnel access.
- Provide access for containment penetrations (mechanical and electrical, piping, etc.).
- Provide an appropriate venting system for the AR to achieve adequate cooling for an indefinite period.

The CES is constructed using SC modular walls for the above and below grade portions. It shares a common Reinforced Concrete (RC) basemat with the CS. Two concentric steel shells form the inner and outer faces of the SC modules, with interconnecting plates providing support. 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.

Below grade, the space between the CES and CS is filled with a Controlled Low Strength Material (CSLM). The AR is ensured to remain leak-tight by use of a flat steel ring that rests on top of the CSLM fill and is welded to the CS shell and the CES wall inner face.

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 is covered by a composite deck lid. The lid has openings for vent pipes to allow water vapor to escape from the AR during a postulated design basis accident, and vertical access for removal of the Steam Generator (SGE), if such replacement becomes necessary.

The design of the CES roof is not yet fully developed. When design details are established, the potential for aircraft impact on the roof will be assessed in line with regulatory expectations.

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

[REDACTED]

Figure 2: Arrangement of the Containment Enclosure Structure, Annular Reservoir and Containment Structure [32]

20.2.2 Containment Structure

The CS is a cylindrical steel containment structure which houses the Reactor Pressure Vessel (RPV), the containment internal structures and several systems. It has an upper torispherical head. The CS is partially and vertically embedded below grade and rests on a steel-lined RC basemat shared with the CES. Above grade, its outer side interfaces with the AR, whereas below grade, its outer side interfaces with the CSLM fill. Figure 2 illustrates the arrangement of the CS.

The CS is designed to perform the following safety functions [33]:

- Provide a leak-tight barrier to prevent or limit the release of radioactive material in all operational states and design basis conditions, throughout the life of the plant.
- House the containment internal structures.
- Provide heat removal and pressure reduction functions as per the PCH.

The CS is designed to perform the following non-safety functions [33]:

- Facilitate controlled access to the SSCs inside the CS while maintaining leak-tightness.
- Facilitate the control of environmental conditions within the containment, for the operation of safety systems, normal operations, and for personnel required to perform operations.

The CS houses the following systems:

- Reactor Coolant System (RCS)
- RPV
- Passive Core Cooling System (PCC) which consists of the following:
    - Primary Decay Heat Removal System (PDH)
    - Secondary Decay Heat Removal System (SDH)
    - Automatic Depressurization System (ADS)
    - Passive Core Makeup Water System (PCM)
- Containment Building Ventilation (CBV)
- Combustible Gas Control System (CGC)
- Spent Fuel Pool (SFP)
- Light Load Handling System (LLH)
- Overhead Heavy Load Handling System (CSH)

The CS is fabricated of low-alloy carbon steel. The CS internal and external surfaces will be coated to prevent corrosion based upon the environment the metal surfaces are exposed to. Above grade, it is reinforced with circular stiffeners welded around its interior. It has a polar crane girder welded to the interior near the upper head and has miscellaneous appurtenances and attachments on both the interior and exterior surfaces.

The CS is supported on a steel baseplate ring backed by a thick layer of reinforced concrete. The baseplate is bolted with long studs to the concrete basemat. Vertical and lateral loads from the CS and the internal structures are transferred to the concrete basemat via the baseplate ring and studs.

The CS is designed to have a clear path above the SGE to the torispherical upper head; the SGE is aligned with the CES vent to facilitate replacement when needed and to support equipment removal during decommissioning.

The CS is equipped with two hatches, the Equipment Hatch (EH) and the Personnel Hatch (PH). The EH is located at the operating floor and permits grade-level access to allow movement of equipment into and out of the containment during refuelling outages. The PH is located below grade level and is used for personnel entry during refuelling outages. Both hatches are designed to maintain containment integrity under design basis conditions, including pressure, temperature and radiation.

Mechanical, electrical and instrument line penetrations are designed to maintain containment integrity under design basis conditions, including pressure, temperature and radiation. The most significant mechanical penetrations of the CS are the main steam piping penetration, main feedwater piping penetration, containment purge piping penetrations, SFP cooling piping penetrations and Residual Heat Removal (RHR) piping penetrations.

20.2.3 Annular Reservoir

The AR is a component of the PCH removal system. It is not a Civil Engineering SSC but it is noted here as it is a significant interface to the CES and the CS. The AR is structurally contained in an approximately [REDACTED] wide annulus between the CES and CS. The CES and CS form the outer and inner wall of the AR, respectively. The AR is present above grade only.

The AR contains a substantial captive body of water, and its primary function is to provide the passive heat sink of the PDH and SDH in the event of a Design Basis Accident (DBA). The AR has sufficient capacity to accept heat from the core, SFP, and containment during a DBA.

Heat from the core may be transferred from the SGE to the AR through the SDH Heat Exchanger (HX). The HX is a component of the SDH, which is a sub-system of the PCC. Natural circulation drives flow through the HX. Additionally, heat from the CS atmosphere may be transferred to the AR by conduction through the 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.

The safety functional requirements of the PCH are presented in Part B Chapter 1 [7] derived from fault studies. The justification of safety functional requirements is under the scope of Part B Chapter 19 [20]. The AR chemistry and chemistry control are discussed in Part B Chapter 23 [24].

20.2.4 Intermediate Building

The IB is an L-shaped, partly underground structure beside the Turbine Building (TB). The IB is designed to be constructed from RC, although its construction may evolve to utilise SC modules similar to the CES. The IB provides a protected area for the Main Steam System (MSS) pipe, the Main Feedwater System (MFS) pipe and the Steam Generator Blowdown

(SGB) Pipes running between the CES and the steam turbine in the TB. The generic SMR-300 features two IBs, each serving one unit.

The IB is divided into two portions based on seismic categorisation. The portion of the IB on the side of the CES houses safety-related components, such as the isolation and relief valves, and therefore is classified as a Seismic Category I structure. The rest of the building houses the remaining length of pipes and a stairway and is classified as a non-seismic structure. Piping which transitions from the seismic to the non-seismic portions of the IB is supported by a seismic restraint at the interface.

The Seismic Category I portion of the IB is completely underground, whereas the non-seismic portion is partially underground. The IB is structurally isolated from the CES by means of a [REDACTED] isolation gap. The IB will be supported on a RC basemat which will be structurally isolated from the basemat of the CES and the TB.

The IB is designed to perform the following safety functions [34]:

- Provide support to the MSS Pipe, the MFS pipe, SGB pipes, Once-Through Steam Generator (OTSG) sampling (liquid and steam), Main Steam Isolation Valve (MSIV) and Main Feedwater Isolation Valve (MFIV) housed inside the IB.
- Provide protection to the safety-related SSCs enclosed within the Seismic Category I portion of the IB against environmental hazards and dynamic effects from postulated pipe break events.
- Provide features (such as rupture bellows and blowout panels) to alleviate any over-pressurization of the IB compartment in a high energy line break event, and to direct it away from personnel.
- Provide structural support to vent piping which releases main steam to the atmosphere (external to the TB and other plan structures) in an event of actuation of safety valves.
- Provide support for the vent lines from the MSS. The vent lines will be anchored into the roof of the Seismic Category I part of the IB to ensure no impact on the MSS lines if the TB was to collapse on the vent stack.
- Provide location for OTSG tube leakage radiation monitors and associated process coolers, as well as main steam line monitoring.
- Provide additional routing for radiation monitoring cooling water lines, fan cooling water lines, sump pump discharge, electrical power cables, shuffle valve drains for the MSS and electrical instrument cabling.
- Provide support for power distribution panels for hot penetration blowers, radiation monitors, electrical heaters, lighting, fan cooler units and sump pumps.

The IB is designed to perform the following non-safety functions [34]:

- Provide metal grating, manways, rungs and removable hatches for equipment maintenance and personnel access.
- Provide an overhead rail and chain hoist system in the Seismic Category I portion of the IB, to allow lifting and removal of valves or their parts.
- Provide sump pumps to avoid flooding in the event of leakage.
- Provide cooling so that any initial conditions or assumptions are maintained, while also allowing for personnel entry and occupancy.

- Provide penetration cooling for process line penetrations in excess of [REDACTED] temperature to ensure that no long-term concrete degradation occurs.
- Provide receptacles for general maintenance and housekeeping purposes.
- Provide general lighting and exit lighting.

The general arrangement of the IB is illustrated in Figure 1 and a plan view is shown in Figure 3.

[REDACTED]

Figure 3: Plan of the Intermediate Building at Below Grade Elevation [35]

20.2.5 Reactor Auxiliary Building

The RAB is a five-story building, with three stories located above grade level and two stories located below grade level. It houses safety-related and non-safety related SSCs. The RAB is designed to be constructed from RC, although its construction may evolve to utilise SC modules similar to the CES.

The RAB combines the functions of a traditional auxiliary building, a traditional fuel handling building and a traditional radioactive waste building into a singular building. It provides two access paths to the containment volume via the EH and the PH. The RAB also contains the Plant Vent Stack. Potentially contaminated effluents would be directed to the plant vent for monitoring before releasing to the atmosphere. CBV effluent would also be routed to the Plant Vent Stack.

The plan arrangement of the RAB is illustrated in Figure 1.

The RAB has both radiologically controlled and non-radiologically controlled areas. It consists of the following areas:

- Auxiliary Systems Area
- Fuel Handling Area
- Remote Shutdown Facility
- Main Control Room (MCR) and essential electrical equipment
- Radioactive Waste Management Area

The Auxiliary Systems Area features auxiliary SSCs that are necessary for the normal operation and shutdown of the plant. It also features safety related SSCs that are required for the safe shutdown of the plant. Adequate shielding and features for removal of radioactive fluids are provided.

The Fuel Handling Area is located at ground elevation. The primary function of the fuel handling area is to provide space for the handling and storage of new fuel and processing of new fuel. It features a Seismic Category I bridge crane to support new and spent fuel transfer activities, and movement of the Low Profile Transporter (LPT) in and out of the CS through the EH.

The RAB includes a Remote Shutdown Facility (RSF), physically separated and located outside of the Control Room Emergency Zone (CREZ).

The MCR area is contained in the CREZ. The MCR provides operators full access to the Plant Safety System (PSS) and Plant Control System (PCS). The MCR area accommodates plant control interfaces and plant operations for monitoring and operation of the plant during normal conditions, and safe maintenance of the plant following a DBA.

The Radioactive Waste Management Area houses equipment and systems for processing radioactive gaseous, liquid, and solid waste and provides facilities for preparation of waste for off-site shipment.

The significant SSCs contained within the RAB are:

- Containment EH and PH entrance
- Class 1E and non-Class 1E electrical equipment and rooms
- Instrumentation and control equipment and rooms
- Mechanical equipment rooms
- Containment mechanical piping penetration areas
- Containment electrical penetration areas
- New fuel vault

Detailed description of the different areas and the SSCs housed within each area is provided in the Design Specification [36].

The RAB is designed to perform the following safety functions [36]:

- Provide protection and separation for Seismic Category I structures, mechanical and electrical equipment located outside the CS:
    - Class 1E Battery Rooms and associated equipment
    - Containment Penetrations/Accessways
    - Safety-related Instrumentation and Control (I&C)
- Provide a control room for plant operators to monitor and operate the plant during normal conditions and to safely shutdown the plant following a DBA.

The RAB is designed to perform the following non-safety functions [36]:

- House SSCs used for the normal operation and shutdown of the plant, such as the RHR System, the Chemical and Volume Control System (CVC), the Primary Sampling System (PSL), the SFP Cooling System, radwaste systems and the Component Cooling Water (CCW) system.
- Provide support for spent fuel removal to interim dry storage.
- Provide support for new fuel receipt and storage.
- Provide Heating, Ventilation and Air Conditioning (HVAC) for all components and structures of the RAB, and containment purging.
- Provide radiation protection for plant personnel from radioactive components, and minimisation of contamination and radioactive waste generation, by incorporating the following design features:
    - Radiation zoning and access control
    - Remotely operated process and instrumentation controls
    - Isolation and decontamination of substantial radiation sources
    - Minimisation of accumulation of radioactive materials in resin and sludge treatment systems, etc.
- Provide suitable environment for personnel and equipment during all phases of plant operation.
- Provide adequate laydown space to facilitate inspection, testing and maintenance of SSCs located within the RAB.
- Provide cooling to all the penetrations into the RAB in excess of [REDACTED] temperature to ensure that no long-term concrete degradation occurs.
- Control, collect, process, handle, store, segregate and dispose of radioactive waste.

- Process and dispose steam generator blowdown if significant radioactivity is detected.
- De-gas reactor coolant in conjunction with the CVC.
- Transport processed gases to the monitored release point.

20.3 CIVIL ENGINEERING CLAIMS, ARGUMENTS, EVIDENCE

This chapter presents the Civil Engineering aspects for the generic SMR-300 and therefore directly supports 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 Part B Chapter 20 taking into consideration the CAE V-model presented in Part A, Chapter 5, Summary of ALARP [37].

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 analysis* 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*, *installation, EIMT* 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 Part B Chapter 9 [10].

Table 1 shows in which sub-chapters of this PSR chapter these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 1: Claims and PSR sub-chapters

Claim No	Claim	sub-chapter
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 AND STANDARDS
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	The civil engineering design utilises good engineering practice to ensure that Civil SSCs are robustly designed to resist design basis loads and load combinations, with appropriate beyond design basis margins.	20.6 DEFENCE IN DEPTH
2.2.11.4	Civil SSCs achieve their design intent through quality manufacturing, installation, examination, inspection, maintenance and testing processes.	20.7 QUALITY MANUFACTURING AND INSTALLATION AND EIMT

Appendix A provides the CAE mapping for Part B Chapter 20, which includes any lower level claims, arguments and evidence needed to support the Claims in the table above. This includes identification of evidence available at PSR v1 and aspects for future development of evidence to support these claims beyond PSR v1.

20.4 CODES AND STANDARDS

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).

This sub-chapter supports Claim 2.2.11.1. Claim 2.2.11.1 has been further decomposed into four arguments to address the claim.

- Civil Engineering SSCs are designed using mature and established nuclear-specific United States (US) and international codes and standards that are considered RGP (A1).
- The relevancy and sufficiency of the US codes and standards and non-compatibility risks with UK RGP have been assessed (A2).
- The design methodology of the generic SMR-300 SC modular concept is developed utilising guidance from relevant nuclear-specific US codes and standards, coupled with physical testing and numerical analysis (A3).
- The CS design concept has been developed using recognised industry standards and has been designed such that it will reliably deliver its passive Safety Functional Requirements (A4).

20.4.1 Civil Engineering Codes and Standards

Claim 2.2.11.1 – A1: The principal codes and standards of the generic SMR-300 are mature and established nuclear-specific US and international codes and standards. They are considered RGP in the UK.

Evidence for Claim 2.2.11.1 – A1 are:

- **HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures** [38]. This report presents the overall design basis of the Civil Engineering SSCs located in the NI. It identifies the codes and standards used for the design of Civil Engineering SSCs.
- **HI-2241295, Structural Analysis and Design Methodology Report** [39]: This report presents the analysis and design methodology for Civil Engineering SSCs. It describes the overall analysis and design approach, the structural analysis methodology for static and dynamic loads, the design methodology and verification and validation.
- **HI-2241210, Aircraft Impact Evaluation Methodology Report** [40]: This report presents the high-level methodology for aircraft impact evaluation of the generic SMR-300. It identifies the applicable codes and standards, the performance requirements and acceptance criteria, the evaluation principles and process, the approach for structural response analysis, induced vibrations and evaluation of fire effects.
- **HI-2241293, Codes and Standards Applicability Report for Civil Engineering** [41]**:** This report identifies the principal codes and standards used in the civil design of the SMR-300. It identifies that the adopted codes and standards are considered RGP by comparison with Office of Nuclear Regulations (ONR) expectations for codes and standards in their Technical Assessment Guides (TAG) and those used by other Requesting Parties during previous GDAs.

The SMR-300 is designed in accordance with the USNRC regulatory guidance and applicable US Code of Federal Regulations (CFR). The standard design of the SMR-300 intends to comply with USNRC requirements set forth in Title 10 CFR Part 50 [42].

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 derived from the classification of SSCs and it is based on USNRC requirements. SSCs of the SMR-300 are classified according to their importance to safety. The classification assigned determines requirements across the lifecycle of SSCs, including compliance to codes and standards.

The applicable USNRC regulations and regulatory Guides (RG) 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 sub-chapter 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. The adopted codes and standards have been developed using decades of nuclear operating experience.

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

Label	Title	Revision/ Date
USNRC 10 CFR Part 50	Domestic Licensing of Production and Utilization Facilities [42]	-
USNRC 10 CFR Part 50, Appendix A	General Design Criteria for Nuclear Power Plants [43]	-
USNRC 10 CFR Part 50, Appendix S	Earthquake Engineering Criteria for Nuclear Power Plants [44]	
USNRC 10 CFR Part 100	Reactor Site Criteria [45]	-
USNRC RG 1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containment) [46].	2020
USNRC 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' [47]	2001
USNRC RG 1.29	Seismic Design Classification for Nuclear Power Plants' [48].	2021
USNRC RG 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components' [49]	2013
USNRC RG 1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants' [50]	2014
USNRC 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 [51]	2007
USNRC 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 [52].	2010
USNRC 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 [53].	2013

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

Reference	Title	Revision/ Date
ACI 349-13	Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary [54]	2013
ANSI/AISC N690-18	Specification for Safety-Related Steel Structures for Nuclear Facilities [55]	2018
ASCE/SEI 4-16	Seismic Analysis of Safety-Related Nuclear Structures [56]	2016
ASME BPVC.III (& XI)	Boiler and Pressure Vessel Code (BPVC) Section III - Rules for Construction of Nuclear Facility Components [57] and subsections: • BPVC Section III, Division 1, Subsection NE – Class MC Components (ASME BPVC.III.1.NE) [58] • BPVC Section III, Subsection NCA, General Requirements for Division 1 and Division 2 (ASME BPVC.III.NCA) [59] • BPVC Section III, Subsection NF – Supports (ASME BPVC.III.NF) [60] • BPVC Section III, Division 2, Subsection CC, Code for Concrete Containments (ASME BPVC.III.2) [61] • BPVC Section XI – Rules for Inservice Inspection of Nuclear Reactor Facility Components, Subsection IWE - Requirements for Class MC and Metallic Liners of Class CC Components of Light Water-Cooled Plants (ASME BPVC.XI.IWE) [62]	2021
NEI 07-13	Methodology for Performing Aircraft Impact Assessments for New Plant Designs [63]	Rev 8P / 2011

The evidence above supports the claim that principal codes and standards are mature and established nuclear-specific US and international codes and standards and are considered RGP by ONR [64] [65].

The SC modular concept for the CES is novel. This is discussed further under Claim 2.2.11.1 – A3.

20.4.2 UK and International Guidance

Claim 2.2.11.1 – A2: The relevancy and sufficiency of the US codes and standards to the UK context have been evaluated. Non-compatibility risks for the application of US codes and standards in the UK context have been identified.

Evidence for Claim 2.2.11.1 – A2 are:

- **HI-2241293, Codes and Standards Applicability Report for Civil Engineering** [41]: Evaluation of the applicability of codes and standards can be necessary even when an applied code or standard is recognised as UK RGP and it is commensurate with the relevant classification, safety functions and required reliability. This is because the approach of applying the code or standard alone may not be consistent with other UK RGP. Consequently, the requirements of the applicable codes and standards may need to be supplemented or modified. This report identifies areas that may be considered for implementation of the adopted codes and standards beyond the GDA, taking into account considerations of the UK context and RGP.

The principal codes and standards applied to the design of nuclear safety related Civil Engineering SSCs are nuclear-specific and are considered RGP by ONR. Furthermore, they have been applied on UK nuclear licensed sites, and earlier successful GDAs. For example, ASCE/SEI 4 Seismic Analysis of Safety-Related Nuclear Structures [56] and ACI 349 Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary [54] are indicated as RGP for seismic analysis of nuclear related civil engineering structures and blast/impact loading respectively in NS-TAST-GD-017 [64].

Several of the principal codes and standards have no equivalent British codes and standards. Examples are ACI 349 [54], ANSI/AISC N690 [55] and ASCE/SEI 4 [56].

HI-2240251 [66] presents the top-level plant design requirements of the SMR-300. The design requirements identified include Holtec SMR-300 Objectives, Electric Power Research Institute (EPRI) requirements for nuclear power plants, as well as other relevant USNRC regulations and guidance.

Potential non-compatibility risks of the application of the adopted codes and standards in the UK context have been identified in HI-2241293 Codes and Standards Applicability Report for Civil Engineering [41]. It is recognised that the potential to use a combination of US design codes and standards combined with local metric material or product codes at the site-specific stage could potentially introduce compatibility risks. Work will be undertaken beyond the GDA to justify such potential incompatibilities. Holtec has produced a Metrication Safety Justification Overview [67]. This sets out the strategy for metrication within the generic SMR-300, and how it will be demonstrated to be ALARP.

The evidence above supports the claim that the relevancy and sufficiency of the US codes and standards to the UK context have been evaluated and that non-compatibility risks in the UK context have been identified.

20.4.3 SMR-300 SC Modular Concept

Claim 2.2.11.1 – A3: The design methodology of the generic SMR-300 steel-concrete modular concept is developed utilising guidance from relevant nuclear-specific US codes and standards, coupled with physical testing and numerical analysis.

Evidence for Claim 2.2.11.1 – A3 are:

- **HI-2241238, Evaluation of Design Concept for SMR-300 Containment Enclosure Structure** [68]: This report presents the rationale for the selection of modular construction for the design of the CES. It sets the basis for substantiating that associated risks can be reduced to ALARP. Furthermore, it identifies the forward actions for developing and substantiating the design and construction methodology. This involves analytical and experimental testing, feedback from USNRC, and development of a detailed construction methodology and schedule.
- **HI-2241293, Codes and Standards Applicability Report for Civil Engineering** [41]: This report discusses the use of the SC modular concept for the design of the CES. This is a more recent form of construction in the UK nuclear industry. There are no UK

or international codes and standards which address specifically this SC modular concept. The design development of the SC modular concept is ongoing based on analytical and experimental testing, informed from relevant codes and standards.

The SC modular design of the generic SMR-300 will utilise relevant guidance from ANSI/AISC N690-18 (incl. Appendix N9) [55], ACI 349-13 [54] and ASME BPVC [57] for, e.g., selection of materials and applicable material design limits, but the design methodology and substantiation will be based on ongoing analytical and experimental studies being undertaken by the Purdue Applied Research Institute (PARI). PARI has significant expertise in advanced computational methods, validation of analytical results with experimental testing, and production of mock-ups to ensure constructability.

The evidence above supports the claim that the design methodology of the SC modular concept will be developed utilising guidance from relevant nuclear-specific US codes and standards, coupled with physical testing and numerical analysis. This is aligned with ONR Safety Assessment Principles (SAP) ECS.4 and ECS.5 [69]:

- ECS.4, Absence of established codes and standards: "*Where there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment, in applications with similar safety significance, should be adopted*".
- ECS.5, Use of experience, tests or analysis: "*In the absence of applicable or relevant codes and standards, the results of experience, tests, analysis, or a combination thereof, should be applied to demonstrate that the structure, system or component will perform its safety function(s) to a level commensurate with its classification*".

Since the design methodology of the SC modular concept is being developed and will not be available within the GDA timeframe, GDA Commitment C_Civi_012 has been raised to provide demonstration that risks are reduced to ALARP. This will be undertaken post-GDA. Refer to sub-chapter 20.8.2.5.

20.4.4 Containment Structure Design Concept

Claim 2.2.11.1 – A4: The Containment Structure design concept has been developed using recognised industry standards and has been designed such that it will reliably deliver its passive Safety Functional Requirements.

Evidence for Claim 2.2.11.1 – A4 are:

- **HI-2250348, Containment Structure System Based View** [70]: The purpose of this report is to present the arguments supporting the demonstration of integrity of the SMR-300 CS. The report supports multiple claims within the PSR by presenting a system-based-view of the CS and summarizing the golden thread of requirements management (from identification to delivery and engineering). This includes a summary of the evidence supporting the design basis assessment against applicable codes and standards.
- **HI-2240159, Design Specification for the Containment Structure** [33]: This document provides a high-level definition of the functional, performance, and safety

requirements forming the basis for the design of the CS for the SMR-300. It provides the classification, safety related and non-safety related functions of the CS.
- **HI-2241293, Codes and Standards Applicability Report for Civil Engineering** [41]: This report evaluates the applicability of codes and standards used in the design of the CS.

As described in sub-chapter 20.2.2, the CS is designed to perform a number of safety functions and non-safety functions. The CS provides a leak-tight barrier to prevent or limit the release of radioactive material in all operational states and design basis conditions, throughout the life of the plant. The CS also acts as part of a safety system to protect against, or mitigate, faults originating within or without the facility. Such faults are captured and assessed further in the scope of Part B Chapter 14 [13], Part B Chapter 15 [14] and Part B Chapter 21 [21] and summarized in the Containment Structure System Based View [70]. The PCH removal system and the role the CS has in the safety functions delivered by this system are discussed further in Part B Chapter 1 [7].

Due to the presence of the AR the failure of the CS can also be a fault initiator, and the resulting indirect consequences are captured in the scope of Part B Chapter 22 [23]. Therefore, the CS can also be considered as a safety-related system due to potential consequences of failure. The scope of Part B Chapter 22 [23], the internal hazards design assessment, will also consider other internal hazards which could lead to CS failure as a consequential event.

Further details on the historic design development of the SMR-300 and the high-level judgment that using passive, as opposed to active, means of achieving a safety function represents relevant good practice, is provided in Part A Chapter 2 [4].

Section 20.4.4.1 sets out the key considerations which have informed the design concept for the CS. This includes aspects such as the choice of a free-standing metal structure, the concept of the AR, the sizing of the CS and the balance of risk between delivery of safety functions and introducing additional hazards to the design. Section 20.4.4.2 then discusses the choice of codes and standards for the CS, such that it will reliably deliver its passive safety functional requirements.

20.4.4.1 Design Concept and Development

The Containment Structure System Based View [70], sets out that the design concept of the CS has considered the following fundamental design inputs:

1. The choice of a free-standing metal containment structure.
2. The concept of the AR and the benefits of the passive nature of the ultimate heat sink.
3. Balancing the overall footprint of the containment structure and the volume/shell thickness needed for the design basis.
4. Design to deliver Safety Functional Requirements and consideration of risks.

These fundamental inputs show how the overall design concept aims to reduce risks to ALARP by preferentially adopting passive engineered safety measures as a means to mitigate design basis accidents.

This approach is judged to represent good practice, and a logical response to global operational experience where failure of active systems and human errors have been shown to be causal or contributing factors in nuclear incidents.

20.4.4.2 Preliminary Classification and Applicable Codes and Standards

As a consequence of the safety significant role of the CS, both as a safety system and as a potential fault initiator, the Containment Structure System Based View [70] identifies a preliminary UK classification for the CS as Class 1. This is because it performs a fundamental role in the transfer of heat to the annular reservoir both through its properties but also its geometry and integrity. This means that the highest level of confidence is needed for the CS to deliver its safety function(s).

The applicable design code for the CS is ASME BPVC Section III, Division 1, Subsection NE (Class MC) [58]. The Containment Structure System Based View [70] justifies the adequacy of this selection, and also considers the applicability of ASME BPVC Section III, Division 1, Subsection NB (Class 1) [71], which has been well established as applicable for Class 1 systems in wet conditions (noting the presence of the AR).

A detailed comparison of ASME BPVC Subsections NE and NB was undertaken to identify differences in approach that may justify the preferential use of Subsection NB [70]. It was identified that Subsection NE is more specific to the application of a containment structure but there are several differences in approach which require justification or explicit incorporation in the design specification to demonstrate applicability of Subsection NE. Class MC provides equivalent assurance as Subsection NB, Class 1.

The conclusion of the review was that use of Subsection NE, Class MC as opposed to Subsection NB in support of a UK Safety Class 1 designation is acceptable, subject to addressing these differences.

It is recognized that with the metal CS surrounded with water in the AR and the exemption from post weld heat treatment (ASME Code Case N841), additional assessments are required on the containment structure to build a robust case for its continued integrity under fault load conditions. GDA Commitment C_Civi_092 has been raised to conduct an assessment on the brittle fracture susceptibility of the containment structure including an assessment into the impact from thermal shock. Refer to sub-chapter 20.8.2.5

It is worth noting that, during GDA Step 2, design changes to the general arrangement of the CES and CS were introduced for the SMR-300 reference design for Palisades. The design changes are documented in HI-2241524 Decision Paper on Containment Structure Design Change [72]. Holtec Britain has assessed these changes as a prospective design change for the generic SMR-300 in line with the Design Management Process [73]**.** The prospective design change has been documented in HI-2250420 UK GDA Prospective Design Change - Containment Structure [74]. Refer to sub-chapter 20.8.2.4.

20.4.5 CAE Summary

This sub-chapter presented the arguments and evidence associated with claim 2.2.11.1. The maturity of evidence provided is generally aligned with the expectations of a PSR.

The SMR-300 Civil Engineering SSCs have been designed using applicable US nuclear codes and standards which are also UK RGP. The adopted codes and standards have been evaluated for their relevancy, adequacy and applicability in the UK context and risks have been identified for consideration beyond the GDA.

The adopted codes and standards have been developed based on several decades of OPEX in the nuclear industry. OPEX from the design, manufacture, construction and operation of nuclear safety related facilities has been used to develop and improve the design codes to ensure they reflect current learning and best practice.

The SC modular concept that will be used for the construction of the CES is novel. As there are no established codes and standards that explicitly apply to the specific concept, a design methodology based on experimental testing and numerical analysis is being undertaken.

The overall design concept of the CS aims to reduce risks to ALARP by preferentially adopting passive engineered safety measures as a means to mitigate design basis accidents. This approach is judged to represent good practice. The CS design concept has been developed using a recognised industry standard, which is UK RGP. The use of ASME BPVC Subsection NE [58] specifically as opposed to Subsection NB [71] in support of a UK Safety Class 1 containment is judged to be acceptable, subject to justifying, post-GDA, differences.

20.5 DESIGN OF CIVIL ENGINEERING SSCs

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

Claim 2.2.11.2 has been further decomposed into three arguments to address the claim.

- Civil Engineering SSC safety and non-safety functions have been identified and the SSCs have been classified taking into consideration the required functions during a Design Basis Event (DBE) (A1).
- Civil Engineering SSCs are designed using USNRC and UK RGP nuclear-specific codes and standards (A2).
- The adopted design methodology for Civil Engineering SSCs within the NI follows UK RGP (A3).

This sub-chapter outlines the analysis methodologies used in the design of SMR-300 Civil Structures. It presents the following:

- **Safety Functions**, which determine the classification of Civil Engineering SSCs.
- **Classification and Seismic Categorisation**, which determine the applicable design codes and standards. As noted in sub-chapter 20.2, Holtec acknowledge the differences in the approach to safety categorisation and classification between USNRC requirements and other national and international standards. This is addressed in Part A Chapter 2 [4]. Holtec have developed an approach for categorisation and classification to meet UK RGP which is defined in HI-2250210, Safety Assessment Handbook (SAH) [30]. The approach defined in the SAH will be applied post-GDA for the Civil Engineering SSCs within the scope of this chapter.
- **Analysis and Design Codes and Standards,** applicable to each Civil Engineering SSC.
- **Analysis and Design Methodology** that will be used to demonstrate that the design meets the requirements of the applicable design codes and standards.

Design management arrangements, including design control processes, are presented in Part A Chapter 4 [6].

20.5.1 Safety Functions

Claim 2.2.11.2 – A1: The Civil SSCs safety and non-safety functions have been identified and the SSCs have been classified taking into consideration the required functions during a DBE.

Evidence for Claim 2.2.11.2 – A1 are:

- **HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures** [38]. This report presents the overall design basis of the Civil Engineering SSCs located in the NI. It defines the safety functional requirements and classification of SSCs.
- **HI-2240310, Design Specification for Containment Enclosure Structure** [31]**:** This document provides a high-level description of the design, function and operation of the

CES. It provides the classification, safety related and non-safety related functions of the CES.

- **HI-2240159, Design Specification for the Containment Structure** [33]: This document provides a high-level description of the functional, performance and safety requirements of the CS. It provides the classification, safety related and non-safety related functions of the CS.
- **HI-2240311, Design Specification for Reactor Auxiliary Building** [36]: This document provides a high-level description of the design, functions and operation of the RAB. It provides the classification, safety related and non-safety related functions of the RAB.
- **HI-2240312, Design Specification for Intermediate Building** [34]: This document provides a high-level description of the design, functions and operation of the IB. It provides the classification, safety related and non-safety related functions of the IB.

Part A Chapter 2 [4] presents the SMR-300 high-level plant functions. The Civil Engineering SSCs with high-level plant functions are summarised in Table 4.

Table 4: 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
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

The safety classification of a function is either safety-related or non-safety related based upon the design function it performs during a 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 (RCPB).
- 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) [75], Title 10 CFR Part 50.67(b)(2) [76], or Title 10 CFR Part 100.11 [77], as applicable.

All other functions are classed as non-safety functions.

The safety and non-safety functions of each Civil Engineering SSC are identified in the design specifications and have been presented in sub-chapter 20.2.

As noted in sub-chapter 20.4.4.2, the safety significant role of the CS, both as a safety system and as a potential fault initiator, is recognised. HI-2250348, Containment Structure System Based View [70] presents the arguments supporting the demonstration of integrity of the CS and supports multiple claims within the PSR by presenting a system-based-view of the CS. A preliminary assessment has identified the UK classification of the CS as Class 1 [70]. Additional work will be undertaken outside the GDA timescales to address differences from UK practice.

20.5.2 Classification and Seismic Categorisation

Claim 2.2.11.2 – A1: The Civil SSCs safety and non-safety functions have been identified and the SSCs have been classified taking into consideration the required functions during a DBE.

Evidence for Claim 2.2.11.2 – A1 are:

- **HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures** [38]. The Design Basis Report for Nuclear Classified Civil Engineering Structures presents the overall design basis of the Civil Engineering SSCs located in the NI. It defines the safety functional requirements and classification of SSCs.
- **HI-2240310, Design Specification for Containment Enclosure Structure** [31]**:** This document provides a high-level description of the design, function and operation of the CES. It provides the classification, safety related and non-safety related functions of the CES.
- **HI-2240159, Design Specification for the Containment Structure** [33]: This document provides a high-level description of the functional, performance and safety requirements of the CS. It provides the classification, safety related and non-safety related functions of the CS.
- **HI-2240311, Design Specification for Reactor Auxiliary Building** [36]: This document provides a high-level description of the design, functions and operation of the RAB. It provides the classification, safety related and non-safety related functions of the RAB.
- **HI-2240312, Design Specification for Intermediate Building** [34]: This document provides a high-level description of the design, functions and operation of the IB. It provides the classification, safety related and non-safety related functions of the IB.
- **HI-2250348, Containment Structure System Based View** [70]: The purpose of this report is to present the arguments supporting the demonstration of integrity of the SMR-300 CS. The report supports multiple claims within the PSR by presenting a system-based-view of the CS and summarizing the golden thread of requirements management (from identification to delivery and engineering). It identifies a preliminary UK class for the CS and includes a summary of the evidence supporting the design basis assessment against applicable codes and standards.

The evidence above supports the claim that safety and non-safety functions for Civil Engineering SSCs have been identified and the SSCs have been classified taking into consideration the required functions during a DBE.

The SMR-300 design approach is based on meeting the applicable sections of Title 10 CFR Part 50 Appendix A General Design Criteria [43] relating to classification, by complying with the applicable requirements of the USNRC Regulatory Guides and NUREG-0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Plants [78]. 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 class and a seismic category. The Holtec SMR classification system uses classes from A to F depending on SSC safety function and quality class. The SMR class system is detailed in Part A Chapter 2 [4].

Furthermore, Generic Design Criteria (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 [44] 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 categorisation 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 [48]. The methodology classifies structures, systems, and components into three categories:

- Seismic Category I
- Seismic Category II
- Non-seismic (NS)

Generally, Seismic Category I applies to both functionality and integrity, whereas Seismic Category II applies only to integrity. Seismic Category I applies to safety-related SSCs and those required to support or protect safety-related SSCs. Seismic Category II applies to SSCs that need to be designed to preclude their structural failure during SSE conditions or interaction with Seismic Category I SSCs. Seismic Category I does not necessarily 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 SSCs are Seismic Category I.

Table 5 provides the classification of the Civil Engineering SSCs within the scope of this PSR chapter and their associated seismic category.

Table 5: SMR Class and Seismic Category of Civil Engineering SSCs

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

20.5.3 Analysis and Design Codes and Standards

Claim 2.2.11.2 – A2: Conformance with USNRC requirements and UK RGP nuclear-specific codes and standards ensures that a reasonably conservative approach is adopted to analyse and design the Civil Engineering SSCs.

Evidence for Claim 2.2.11.2 – A2 are:

- **HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures** [38]: This report presents the overall design basis of the Civil Engineering SSCs located in the NI. It identifies the safety functional requirements of SSCs, the SSC classification, the applicable USNRC requirements, the adopted codes and standards, and the design specifications of individual SSCs and their design bases. It identifies the principal codes and standards, which are UK RGP.
- **HI-2241295, Structural Analysis and Design Methodology Report** [39]: This report presents the analysis and design philosophy of Civil and Engineering SSCs. It presents the analysis and design methodology and the adopted structural analysis and design codes and standards, which are UK RGP.
- **HI-2241210, Aircraft Impact Evaluation Methodology Report** [40]: This report presents the aircraft impact evaluation methodology of the generic SMR-300. It identifies the applicable standard, which is UK RGP.

The principal Civil Engineering design codes and standards for SSCs within the scope of this are summarised in Table 6 against the SSCs they are applied to.

Table 6: Principal Codes and Standards applied to Civil Engineering SSCs

SSC/Technical topic	SMR-300 Codes and Standards
Containment Structure (CS)	• ASME BPVC Section III, Division 1, Subsection NE [58] • ASME BPVC Section III, Subsection NCA [59] • ASME BPVC Section III, Subsection NF – Supports [60]
Containment Enclosure Structure (CES) SC modular concept	Ongoing development of design methodology based on physical testing and numerical analysis.
Basemat	ACI 349-13 [54] ASME BPVC III-2 [61]
Reinforced concrete structures	ACI 349-13 [54]
Steel structures	ANSI/AISC N690-18 [55]
Seismic analysis	ASCE/SEI 4-16 [56]
Aircraft impact evaluation	NEI 07-13 [63]

Sub-chapter 20.4 identifies the relevant USNRC requirements and the design codes and standards adopted for the generic SMR-300 design. The aforementioned documents also identify the adopted codes and standards, which are established, mature, internationally recognised, nuclear-specific, and UK RGP. The adopted codes and standards have been developed using decades of nuclear operating experience. Nuclear-specific codes and standards are inherently conservative. They reflect the required reliability of nuclear safety-related SSCs commensurate with their safety significance, and the functions that they perform.

An exception for which an established code or standard does not currently exist is the SC modular concept of the CES. The design methodology of the SC modular concept is currently in development utilising guidance from relevant nuclear-specific US codes and standards, together with physical testing and numerical analysis undertaken in partnership with PARI. This is aligned with ONR SAPs ECS.4 ‘Absence of established codes and standards’ and ECS.5 ‘Use of experience, tests or analysis’ [69] and is addressed by argument 2.2.11.1 – A3.

The evidence above supports the claim that the adopted codes and standards conform with USNRC requirements and UK RGP to ensure a reasonably conservative approach for the analysis and design of Civil Engineering SSCs.

20.5.4 Analysis and Design Methodology

Claim 2.2.11.2 – A3: The adopted design methodology for the Civil SSCs within the Nuclear Island follows the UK RGP. It includes, but is not limited to, the rationale of material selection, safety classification, safety functional requirements for each of the Civil SSCs, load combinations and their acceptance criteria, structural modelling, seismic analysis, aircraft impact evaluation and verification and validation of numerical simulations.

Evidence for Claim 2.2.11.2 – A3 are:

- **HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures** [38]: This report presents the overall design basis of the Civil Engineering SSCs located in the NI. It defines the safety functional requirements and classification of SSCs, the adopted codes and standards, material properties, analysis and design methodology, design loads and load combinations and design acceptance criteria.

- **HI-2241295, Structural Analysis and Design Methodology Report** [39]: This report presents the analysis and design methodology for Civil Engineering SSCs. It describes the overall analysis and design approach, the structural analysis methodology for static and dynamic loads, the design methodology and verification and validation.
- **HI-2241210, Aircraft Impact Evaluation Methodology Report** [40]: This report presents the high-level methodology for aircraft impact evaluation of the generic SMR-300. It identifies the applicable codes and standards, the performance requirements and acceptance criteria, the evaluation principles and process, the approach for structural response analysis, induced vibrations and evaluation of fire effects.
- **HI-2240314, Civil Structural Design Basis for Containment Structure Internals and Containment Enclosure Structure** [79]: This report provides the methodology, parameters, applicable codes and standards, acceptance criteria, design loads and design considerations for the structural design of the CES.
- **HI-2240315, Civil Structural Design Basis for RAB** [80]: This report provides the parameters, applicable codes and standards, acceptance criteria, design loads and design considerations for the structural design of the RAB.
- **HI-2240316, Civil Structural Design Basis for Intermediate Building** [81]: This report provides the methodology, parameters, applicable codes and standards, acceptance criteria, design loads and design considerations for the structural design of the IB.

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

20.5.4.1 Analysis and Design Approach

The NI includes a deeply embedded reactor building in proximity with other structures. The interaction of structures with the surrounding subgrade is an important consideration for accurately evaluating the response of the NI structures under static and dynamic loads. For this reason, the analysis approach of the generic SMR-300 involves Soil-Structure Interaction (SSI) analysis.

The key aspects of the analysis and design approach are:

- A two-step approach is adopted for the analysis of structures.
- The SSI analysis utilises a state-of-the-art integrated Finite Element (FE) model of the NI structures.
- Potential geometric nonlinearities at the soil-structure contact interface and soil material nonlinearities in the region in proximity with the embedded structures are captured in the SSI analysis.
- Post-processing of the SSI analysis is based on an in-house developed automated approach.
- Sensitivity studies and verifications and validation are undertaken to provide confidence that the results are accurate and appropriate.

The first step of the analysis considers SSI of the NI structures with the surrounding soil. For this step, structural modelling needs to be sufficiently detailed to predict the global behaviour. An integrated three-dimensional (3D) FE model comprising all structures of the NI and a sufficiently large soil region has been developed using LS-DYNA.

System-level responses (e.g., floor acceleration time-histories, in-structure response spectra, etc.) are obtained from the first step SSI analysis and used as input for the second step. The second step considers a decoupled soil-structure problem which consists of structures and/or parts of structures without the soil. For the second step, the discretization and detail of structural finite element models are refined to predict accurately local behaviour.

The results of the second step analyses are used to determine structural demands, such as internal forces, displacements, stresses, accelerations, and reactions. Code-based design checks are undertaken using built-in design tools in FE software packages or in-house developed automated routines.

Verification and validation will be undertaken to provide confidence that the developed computational models produce sufficiently accurate results and are suitable for the intended purpose.

The analysis and design approach workflow is illustrated in Figure 4.

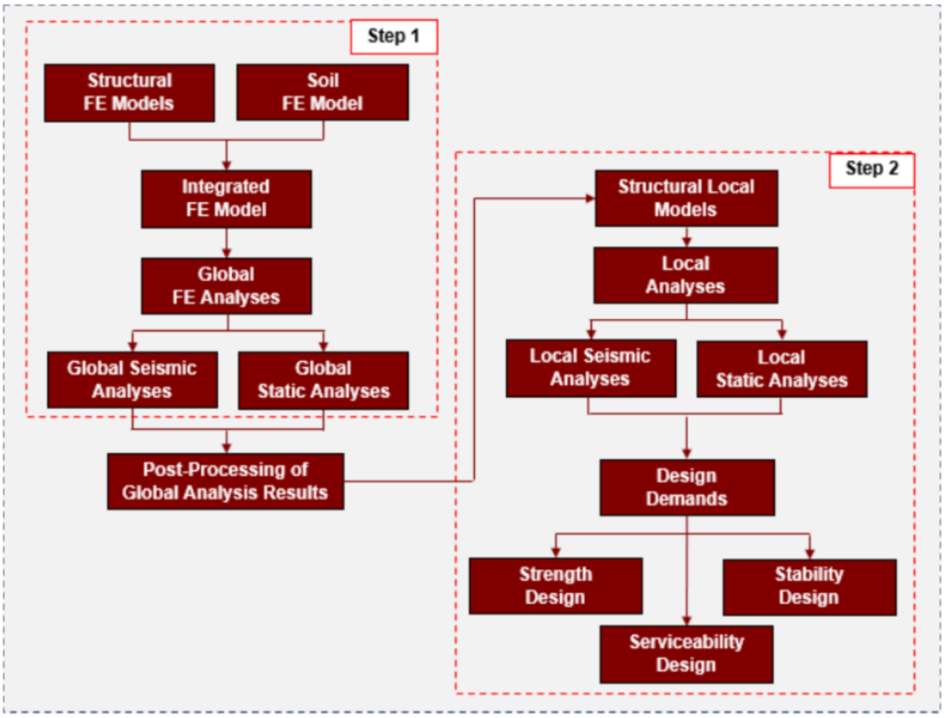

Figure 4: Analysis and Design Approach of SMR-300 Nuclear Island Civil Engineering Structures [39]

20.5.4.2 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 and are identified in the relevant design specifications [31] [33] [34] [36].

The steel containment vessel is made of American Society for Testing and Materials (ASTM) [REDACTED] steel. The materials used for the construction of other structures meet the applicable requirements of ACI 349 [54] and ANSI/AISC N690 [55]. The design of the SC modular concept is in ongoing development.

Exclusion of the exterior surface of the CS and the interior surface of the CES from the chemical environment of the AR is achieved through selection of coatings [33] [31]. Application of the coating systems will be in accordance with PS-8002-0020 SMR-300 Specification – Protective Coatings [82].

Choice of the coating for the CS takes into consideration the general functional requirements of the CS which includes heat transfer in addition to the expected functions and compatibilities required for a coating. The choice of coating for the CS is evaluated in HI-2250348 Containment Structure System Based View [70]. This evaluation concludes that although the design standard adopted for the CS coating is proportionate to the classification of the CS and delivers the required reliability and assurance, the choice of coating is not currently well defined and interfaces with the chemistry controls for the AR. GDA Commitment C_Civi_093 has been raised to review the approach to material degradation, coating arrangements, and the ALARP position for the CS post-GDA. Refer to sub-chapter 20.8.2.5.

20.5.4.3 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. The loads and load combinations used for the analysis and design of Civil Engineering SSCs are provided in the relevant design specifications [31] [33] [34] [36], design basis documents [79] [80] [81] and the Design Standard for Basic Civil Structural Requirements According to Seismic Class [83] .

Normal operational loads are the maximum loads induced during normal plant operation and shutdown. The external hazards and their combinations considered in the generic SMR-300 design are derived in Part B Chapter 21 [21]. The internal hazards and their combinations considered in the generic SMR-300 design are derived in Part B Chapter 22 [23].

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

20.5.4.4 Structural modelling

Civil Engineering structures are analysed using 3D FE modelling techniques for both static and dynamic analyses using industry-standard software such as LS-DYNA and ANSYS. 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.

20.5.4.5 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 [50], anchored at a Peak Ground Acceleration (PGA) of [REDACTED] in each orthogonal direction. Notably, the SDRS are anchored at the base of the CES at an elevation of almost [REDACTED] below grade.

Refer to the Generic Site Envelope Report [22] for commentary comparing the SMR-300 SDRS against the spectra derived to define the SMR-300 Great Britain (GB) Generic Site Envelope (GSE).

A time-domain nonlinear SSI analysis methodology is employed to assess the seismic responses of the Seismic Category I structures and neighbouring structures within the SMR-300 plant boundary.

The outcome of the SSI analysis provides input to subsequent structural evaluations, encompassing the design of Civil Engineering SSCs, as well as defining seismic loads for assessing safety-related equipment and their supporting structures. The SSI analysis will provide In-Structure Response Spectra (ISRS) at different locations of the safety-related structures, including the CES. The application of response spectra for structural qualification of the CES will be at its foundation level, while accounting for the presence of soil and other structures surrounding it. This is consistent with the expectations of NUREG-0800 Chapter 3, which is recognised as good practice by NS-TAST-GD-017 Annex 1 [64].

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 AR and in the spent fuel pool is modelled explicitly using solid elements associated with a 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 concrete 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 LS-DYNA model is verified against an equivalent SASSI model. This verification is undertaken for low seismic intensities since SASSI is linear elastic. For this verification, the contact interfaces between the embedded parts of the plant and the subgrade are assumed to be perfectly bonded.

Sensitivity analyses are undertaken to address the variability and uncertainties related to the following:

- Soil properties (upper bound, best estimate, lower bound).
- Groundwater level.
- Mesh discretization.
- Concrete cracking.

The dynamic FE models are developed, verified and validated using a phased approach. The seismic analysis methodology is detailed in the Structural Analysis and Design Methodology Report [39].

20.5.4.6 Aircraft Impact Assessment

The aircraft impact evaluation methodology is provided in the Aircraft Impact Evaluation Methodology Report [40]. The SMR-300 generic design considers aircraft impact according to the guidance adopted from the USNRC. For any new reactor designed after July 13, 2009, Title 10 CFR Part 50.150 [84] 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 US.

The following general requirements apply for the generic SMR-300 design following aircraft impact [85]:

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

The following structures shall be demonstrated that they are able to continue to perform their safety functions following the external event:

- CES and CS.
- MCR and RSF areas of the RAB.

Missile effects on important to safety SSC shall be considered. 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. 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 Part B Chapter 21 [21], the Aircraft Impact Safety Case Strategy Report [85] and the Aircraft Impact Evaluation Methodology Report [40].

It is worth noting that Design Challenge HI-2250436-R0.0 identifies that the current CES roof and aspects of the RAB design may require additional justification to meet UK expectations for resilience against aircraft impact due to US / UK regulatory differences. GDA Commitment C_Exte_066 is raised in Part B Chapter 21 [21] to progress this Design Challenge through the Design Management process [73] to completion. Target for Resolution is Issue of Pre-Construction SSEC.

20.5.4.7 Design Methodology

The design methodology is provided in the Structural Analysis and Design Methodology Report [39].

Results obtained from the local (second step) analyses (e.g., internal forces, stresses, strains, displacements) are used to calculate the design demands on structures.

The design demands are obtained by:

- Implementing the relevant load case combinations derived from the applicable design codes and standards reported in the Design Basis Report [38]. The local analyses are linear elastic; therefore, the principle of superposition is adopted for determining the design demands. The combined effects are obtained by the algebraic sum of the effects resulting from each load case.
- Enveloping the results across the soil profiles.

The seismic demand on structural members is computed by spatially combining the results in the three orthogonal directions using either the Square Root of the Sum of the Squares (SRSS) method or the 100-40-40 rule as described in Section 4.6 of ASCE/SEI 4-16 [56].

The design demands are used to undertake the following, based on the requirements of the applicable codes and standards:

- Strength design.
- Stability checks.
- Serviceability design.

Design calculations are undertaken using common engineering tools, e.g., PTC Mathcad, Microsoft Excel, or specific design modules in FE software packages. The design outcome is expressed in terms of Utilisation Ratios (UR). Appropriate margins of URs from unity are considered.

The process for undertaking design checks is depicted in Figure 5.

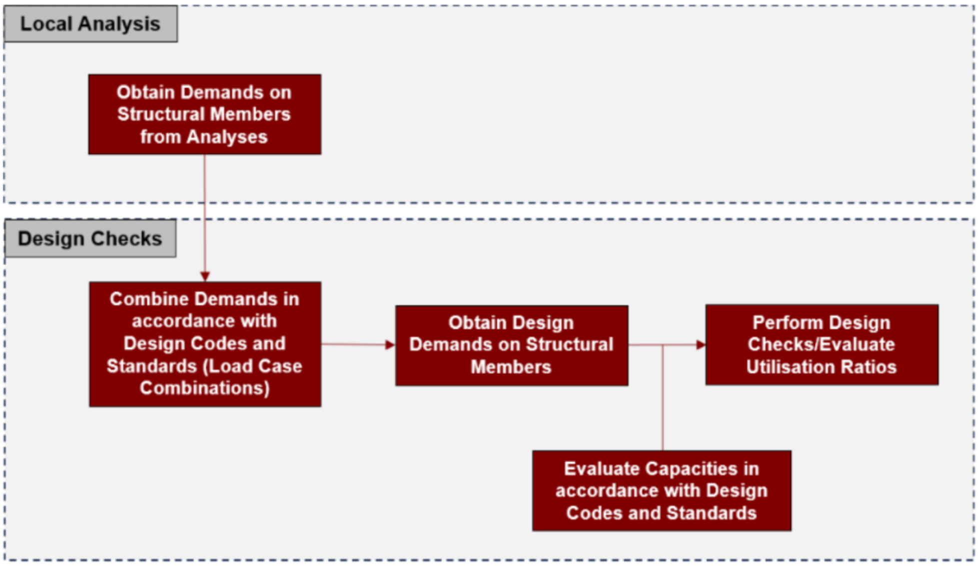

Figure 5: Process for Undertaking Design Checks of the SMR-300 Civil Engineering Structures [39]

20.5.5 CAE Summary

This sub-chapter presented the arguments and evidence associated with claim 2.2.11.2. The maturity of evidence provided is aligned with the expectations of a PSR.

The SMR-300 Civil Engineering SSCs have been designed using best practice analysis methodologies which include:

- Design to applicable US nuclear codes and standards and UK RGP ensures that a reasonably conservative approach is adopted and any differences between US codes and UK RGP identified as potential deviation risks.
- A robust structural analysis and design methodology has been adopted which reflects UK RGP.

In addition, the safety and non-safety functions of civil SSCs have been defined and their classification determined taking into consideration the required functions during a DBE. The safety and non-safety function and classification of the SMR-300 aligns and complies with USNRC requirements. Holtec acknowledge the differences in the approach to safety categorisation and classification between USNRC requirements and other national and international standards. This is addressed in Part A Chapter 2 [4]. Holtec have developed an approach for categorisation and classification to meet UK RGP, as defined in HI-2250210 Safety Assessment Handbook [30]. The approach to meet UK RGP will be applied after GDA Step 2.

20.6 DEFENCE IN DEPTH OF CIVIL ENGINEERING

Claim 2.2.11.3: The civil engineering design utilises good engineering practice to ensure that Civil SSCs are robustly designed to resist design basis loads and load combinations, with appropriate beyond design basis margins.

Claim 2.2.11.3 has been further decomposed into three arguments to address the claim:

- The civil engineering design uses design and analysis methods which are considered RGP, using conservative material properties and load cases (A1).
- The civil engineering structures are analysed to demonstrate that the design is robust during all design basis load cases (A2).
- The civil engineering structures are analysed to demonstrate no cliff-edge effects from Beyond Design Basis (BDB) hazards (A3).

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 SAPs for Nuclear Facilities [69] 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

Claim 2.2.11.3 – A1: The civil engineering design of the generic SMR-300 is undertaken using design and analysis methods which are considered RGP, using conservative material properties and load cases (including load combinations), that meet the requirements of nuclear-specific RGP codes and standards.

Evidence for Claim 2.2.11.3 – A1 are:

- **HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures** [38]: This report defines the design and analysis methods, material properties and load cases. The loads, load cases and load combinations at GDA are the same as those defined for the US SMR-300 design.
- **HI-2241295, Structural Analysis and Design Methodology Report** [39]: This report presents the analysis and design methodology for Civil Engineering SSCs. It also presents the methodology for sensitivity studies and verification and validation. The analysis and design methodology are consistent with the requirements of RGP nuclear specific codes and standards and aligned with international guidelines. A state-of-the-art SSI analysis method is used for the NI Civil Engineering SSCs, consistent with requirements of RGP codes and standards.

The design of the generic SMR-300 utilises appropriate, mature, established and internationally recognised nuclear safety related codes and standards. This is supported by the USNRC endorsement of these codes and standards, and their adoption as mandatory in their prescriptive regime. Furthermore, the adopted codes and standards are recognised as RGP by ONR. Nuclear codes and standards are generally inherently conservative, through conservative assumptions and safety margins in material properties and safety factors, loads and load combinations and overall design requirements.

The generic SMR-300 codes and standards have been developed and evolved over several decades of operational experience and regulatory scrutiny, to enhance the safety and reliability of engineering SSCs, incorporate lessons learned and adapt to new technologies, engineering and regulatory advancements.

20.6.2 Design Basis Accident Analysis

Claim 2.2.11.3 – A2: For faults within the design basis, the civil engineering structures are analysed to demonstrate that the design is robust during all design basis load cases, including natural and human-induced internal and external hazards.

Evidence for Claim 2.2.11.3 – A2 are:

- **HI-2240345, Part B Chapter 14, Safety and Design Basis Accident Analysis** [13]: Part B Chapter 14 presents the Claims, Arguments, and Intended Evidence to demonstrate that the generic SMR-300 design and operation are tolerant to faults and that applicable UK safety targets are met. It is the overarching topic of the SSEC for design basis accident analysis.
- **HI-2241323, Preliminary Fault Schedule Report** [86]: This report records the faults and protection systems of the generic SMR-300.
- **HI-2240069, Generic Site Envelope Report for SMR-300 UK GDA** [22]: The GSER defines the design basis external hazards relevant to the SMR-300 on the UK generic site.
- **HI-2241054, Internal and External Hazards Combined Hazards Methodology** [87]: This report presents the methodology for identification, categorisation and screening of internal and external hazard combinations. It also considers coincidental external hazards, combinations of three or more external hazards and combinations of external hazards with internal hazards.

- **HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures** [38]: This report identifies the design basis load cases and load combinations for static and dynamic loading and natural and human-induced internal and external hazards.

DBAA within the generic SMR-300 SSEC is presented in Part B Chapter 14 [13], which aims to demonstrate that there is a robust methodology for the identification and assessment of fault conditions.

The Fault Schedule is a recognised tool within the UK nuclear regulatory system. It details the relevant faults and postulated initiating events that cover the design basis and design extension conditions that will be assessed within the safety analysis. During GDA, the fault schedule will be preliminary to reflect that the design is under development.

The evidence above supports the claim that civil engineering structures are analysed to demonstrate that the design is robust during all design basis load cases, including natural and human-induced internal and external hazards.

20.6.3 Beyond Design Basis Accident Analysis

Claim 2.2.11.3 – A3: The generic SMR-300 Beyond Design Basis Accident (BDBA) analysis considers internal reactor hazards, natural and human-induced internal and external hazards. The BDBA analysis will quantify margins in terms of hazard magnitude or SSC withstand.

Evidence for Claim 2.2.11.3 – A3 are:

- **HI-2240346, Part B Chapter 15, BDBA, Severe Accidents Analysis and Emergency Preparedness** [14]: Part B Chapter 15 presents the Claims, Arguments and Intended Evidence for BDBA, Severe Accident Analysis and Emergency Preparedness that underpins the design of the generic SMR-300.
- **HI-2240350, Part B Chapter 21 External Hazards** [21]: Part B Chapter 21, together with its associated Step 2 deliverables, provides a list of the BDB external hazards that will be considered in the detail design of civil SSCs.
- **HI-2240351, Part B Chapter 22 Internal Hazards** [23]: Part B Chapter 22, and its associated Step 2 deliverables, do not currently provide a list of the BDB internal hazards that will be considered in the detail design of civil SSCs. A list of BDB internal hazards will be derived following further work after GDA Step 2.
- **HI-2241236, Beyond Design Basis Strategy for External Hazards Report** [88]: This report presents the strategy for demonstrating absence of cliff-edge effects from external hazards, and provides input to Part B Chapter 21 [21].

BDBA analysis considers the design response to a hazard whose magnitude is above the design basis and aims to demonstrate an acceptable margin to failure. The BDBA of the generic SMR-300 considers 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.

A preliminary BDB assessment has been conducted for the external hazards considered in the GB GSE for the generic SMR-300 GDA. The results are summarised in the Beyond Design

Basis Strategy for External Hazards Report [88]. It is worth noting that for the seismic external hazard, which is often a bounding external hazard for the design basis of civil structures, the margins present in the GDA reference design provide confidence that that there is sufficient margin to preclude the possibility of cliff-edge effects. However, further BDB evaluation will be required at the site-specific stage when maturity of the design allows for site-specific seismic assessments.

Furthermore, preliminary BDB evaluations for extreme wind, tornadic wind, tornado-generated missiles, extreme rainfall and snow found that the GDA reference design parameters sufficiently bound the GB GSE parameters such that the margin is considered sufficient to preclude the possibility of cliff-edge effects [88]. Further BDB evaluations are required beyond GDA to assess margins from other external hazards (extreme ambient air temperature, humidity, external flooding, etc.).

BDB internal hazards have not been identified at GDA Step 2, but it is expected that a number of overpressure hazards due to hydrogen explosion or steam release and their impact on building structures will require assessment. This work will be undertaken after GDA Step 2.

The evidence above supports the claim that BDB hazards have been considered in the generic SMR-300 design and further work will be undertaken after GDA Step 2 to quantify the margins in the design.

20.6.4 CAE Summary

This sub-chapter presented the arguments and evidence associated with claim 2.2.11.3. The maturity of evidence provided is aligned with the expectations of a PSR.

The SMR-300 Civil Engineering SSCs have been designed using design and analysis methods which are considered RGP, using conservative material properties and load cases. The civil engineering structures are analysed to demonstrate that the design is robust during all design basis load cases. The civil engineering structures are analysed to demonstrate no cliff-edge effects from BDB hazards.

20.7 QUALITY MANUFACTURING AND EIMT

Claim 2.2.11.4: Civil SSCs achieve their design intent through quality manufacturing, installation, examination, inspection, maintenance and testing processes.

Claim 2.2.11.4 has been further decomposed into two arguments to address the claim:

- The SMR-300 maintains high quality standards throughout the lifecycle of the SMR-300 (A1).
- The SMR-300 is designed taking into account EIMT, to ensure that civil SSCs will continue to meet their design intent throughout the lifetime of the SMR-300 (A2).

This sub-chapter provides information, the claims, arguments and evidence on the following aspects associated with Civil Engineering SSCs:

- Quality Assurance (QA)
- EIMT
- Construction and Fabrication

20.7.1 Quality Assurance

Claim 2.2.11.4 – A1: The generic SMR-300 project has controls in place to ensure that the design of the SMR-300 maintains high quality standards throughout the lifecycle of the SMR-300. Quality assurance requirements for Civil SSCs are established in design specifications based on appropriate nuclear-specific codes and standards.

Evidence for Claim 2.2.11.4 – A1 are:

- **HI-2240335, Holtec SMR-300 GDA PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance (MSQA)** [6]**:** Part A Chapter 4 presents the arrangements for the lifecycle MSQA of the generic SMR-300. The lifecycle MSQA arrangements are the set of quality arrangements that allow safety, security, safeguards and environmental protection matters to be managed throughout the lifecycle of the plant.
- **HI-2240310, Design Specification for Containment Enclosure Structure** [31]**:** This document provides a high-level description of the design, function and operation of the CES. Additionally, it provides requirements for the design, fabrication, construction, operation and maintenance of the CES in compliance with appropriate codes and standards.
- **HI-2240159, Design Specification for the Containment Structure** [33]: This document provides a high-level description of the functional, performance and safety requirements of the CS. Additionally, it provides requirements related to the design and fabrication of the CS in compliance with applicable codes and standards, and regulatory requirements.
- **HI-2240311, Design Specification for Reactor Auxiliary Building** [36]: This document provides a high-level description of the design, functions and operation of the RAB. Additionally, it provides requirements for design, fabrication and construction, maintenance, inspection, testing and surveillance.

- **HI-2240312, Design Specification for Intermediate Building** [34]: This document provides a high-level description of the design, functions and operation of the IB. Additionally, it provides high-level requirements for fabrication, inspection, maintenance and testing. Construction requirements will be developed later during detailed design.

The overarching QA arrangements for maintaining high quality standards throughout the lifecycle of the SMR-300 are described in Part A Chapter 4 [6].The QA requirements for civil structures can be found in the associated design specifications for the key buildings and structures listed above.

The design specifications include requirements for:

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

QA requirements are generally imparted from the applicable design codes and USNRC guidance, e.g., 10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors [89].

It is worth noting that the plant design philosophy and high-level requirements of the SMR-300 are provided in the SMR-300 Top Level Plant Requirements [66], refer to Part A Chapter 4 [6]. The SMR-300 Top Level Plant Requirements [66] provides the guiding principles for plant design and related actionable requirements that the design specifications above are required to align with. The plant design philosophy for safety, performance and constructability, utilising guidance from EPRI, has had the largest influence on the configuration of the design and the design requirements which QA can be assured against.

The evidence above supports the claim that the SMR-300 project has controls in place to ensure that the design of the SMR-300 maintains high quality standards throughout the lifecycle of the SMR-300 and that QA requirements for civil SSCs are defined.

20.7.2 Examination, Inspection, Maintenance and Testing

Claim 2.2.11.4 – A2: The generic SMR-300 is designed taking into consideration EIMT, to ensure that civil SSCs will continue to meet their design intent throughout the lifetime of the SMR-300, including provision for appropriate access, maintenance and inspection. An effective EIMT programme will be in place throughout the lifetime of the facility.

Evidence for Claim 2.2.11.4 – A2 are:

- **HI-2240340, Holtec SMR-300 GDA PSR Part B Chapter 9 Description of Operational Aspects and Conduct of Operations** [10]**:** Part B Chapter 9 presents the Holtec SMR-300 approach to plant operations and the undertaking of operations. It presents the high-level plant procedures for EIMT and managing ageing and degradation.

- **HI-2240310, Design Specification for Containment Enclosure Structure** [31]**:** This document provides a high-level description of the design, function and operation of the CES. Additionally, it provides requirements for inspection, testing and maintenance in compliance with appropriate codes and standards.
- **HI-2240159, Design Specification for the Containment Structure** [33]**:** This document provides a high-level description of the functional, performance and safety requirements of the CS. Additionally, it provides requirements for inspection, testing and surveillance in compliance with applicable codes and standards, and regulatory requirements.
- **HI-2240311, Design Specification for Reactor Auxiliary Building** [36]**:** This document provides a high-level description of the design, functions and operation of the RAB. Additionally, it provides requirements for maintenance, inspection, testing and surveillance.
- **HI-2240312, Design Specification for Intermediate Building** [34]**:** This document provides a high-level description of the design, functions and operation of the IB. Additionally, it provides high-level requirements for inspection, maintenance and testing.

The operational design life of the generic SMR-300 design is 80 years. The operational design life refers to the expected period of operation during which Civil Engineering SSCs are intended to function safely and effectively without requiring major repairs or replacement, provided that proper maintenance is undertaken. Civil Engineering structures will be required to perform their respective safety functions beyond the 80-year operational design life, to account for construction and decommissioning. An effective EIMT programme shall be in place throughout the full lifetime of the facility.

Engineering structures will be designed according to best industry practice in terms of material properties, supplier guarantees, loading conditions, environmental exposure, usage requirements, etc., along with establishing an appropriate EIMT plan to enable the SMR-300 plant to operate and demonstrate through-life reliability for at least 80 years, with anticipated maintenance but without major repairs.

The overarching EIMT approach for the SMR-300 is described in Part B Chapter 9 [10]. EIMT requirements for each safety significant Civil Engineering SSC are defined in the respective design specification documents listed above for the key buildings and structures. This includes requirements for pathways/platforms for personnel access to allow EIMT activities to take place for managing ageing and degradation.

The SMR-300 Top Level Plant Requirements [66] provides the high-level requirements used for the design of the generic SMR-300. It includes operability and maintainability requirements that are applicable for all SMR-300 SSCs, including Civil Engineering SSCs, and informs the design development of the plant. EIMT requirements are also informed by the SMR-160 Design Standard for Human Factors: Maintenance, Inspection and Testing [90], which remains applicable to the design of the SMR-300.

A SmartPlant 3D model of the SMR-300 is employed to inform the ongoing design and layout development of the plant and facilitate EIMT considerations. This model has been developed and is continuously updated in compliance with SmartPlant Standards [91], which provides the standard for modelling, communication and coordination of SmartPlant 3D in the design of the SMR-300.

In general, EIMT requirements of reinforced concrete and steel structures are in accordance with the requirements of ACI 349 [54] and AISC N690 [55], respectively. EIMT requirements of the CS are in accordance with Article NE-6000 of ASME BPVC Section III, Division 1, Subsection NE [92] (inspection and testing, including pre-operational structural proof test), Subsection IWE of ASME BPVC Section XI [62] (preservice examination, in-service inspection, repair and replacement) and 10 CFR Part 50 Appendix J [89] (leakage testing). EIMT requirements of the CES SC modules are in development.

Detailed EIMT requirements and procedures will be developed later as the design matures further. Part B Chapter 9 [10] identifies EIMT activities that will be conducted beyond the GDA. The EIMT and Ageing and Degradation programme will be developed as the generic SMR-300 design evolves. Although the Ageing and Degradation programme has not yet been formalized, ageing and degradation mechanisms are being identified and mitigated using the Holtec Product Development Design Control Process (DCP) [93]. The DCP is certified to the ASME NQA-1 QA standard for supply items or services that provide a safety function for nuclear facilities.

EIMT considerations are captured using the DCP and recorded in HI-DOC, Holtec's document management system. All documents and drawings, regardless of complexity, have a drafting review period prior to being released for analysis or calculations. A Technical Sponsor determines the multi-disciplinary reviews required. The reviews include Holtec subject matter experts. Review comments and markups are recorded in HI-DOC. Comments are provided to the Technical Sponsor for review and approval. The Technical Sponsor works with the Preparer to ensure that all EIMT comments are correctly incorporated in the design documentation and drawings.

Furthermore, the Holtec SMR-300 EIMT development approach takes into consideration the Construction (Design and Management) 2015 Regulations (CDM 2015) [94] expectations, by addressing health and safety during the design development. The following two documents evidence how Holtec Britain will discharge their duties as a CDM 2015 Designer, both documents are also summarised in Part B Chapter 12 [17]:

- Holtec Britain CDM Strategy [95]
- SMR-300 GDA Nuclear Site Health and Safety Management System [96].

With regards to the decommissioning phase, further work as normal business will include detailing a decommissioning plan. In preparation of the Post Operational Clean-Out (POCO) stage of the plant, the decommissioning plan will be informed by the operational history and a baseline survey of the plant. Delayed decommissioning would not benefit greatly from radioactive decay as the primary isotopes of concern include longer lived ones. A detailed decommissioning waste inventory shall be developed beyond GDA which shall inform risk assessments for decommissioning. An ALARP case can be made for prompt decommissioning of the SMR-300.

20.7.3 Construction and Fabrication

The overarching approach for the construction of the generic SMR-300 is provided in Part B Chapter 25 [25].

A significant portion of the reactor building, which includes the CES, is deeply embedded. This is primarily driven by the passive safety design philosophy of the plant, but also presents other benefits such as reduced exposure to external hazards, improved seismic stability and potentially improved radiation shielding. Deeply embedded structures can pose challenges during the different phases of a project; for example, health and safety during excavation, confined spaces, inaccessible areas for EIMT during operation of the plant, etc. Construction complexities due to deep excavation can lead to delays and increased cost and are heavily influenced by site-specific conditions (e.g., groundwater levels, type of ground material, etc.).

Although there is no precedent of a deeply embedded reactor building having previously been constructed in the UK, deep excavations have been undertaken for other UK nuclear facilities which are regulated by ONR. One of the most notable examples is Hinkley Point C, which involved excavations up to 35 m deep in bedrock, a 46 m deep shaft and outfall/intake tunnels up to 33 m below the seabed. Deep excavations have also been carried out at Sellafield, where underground facilities have been built for waste storage and reprocessing.

Holtec will ensure that proper measures are employed to mitigate any conventional health and safety risks associated with the deeply embedded CES. For example, the Nuclear Site, Health & Safety team of Holtec Britain has issued the two following documents to the ONR which set out a proposal to demonstrate that CDM 2015 [94] requirements are understood and will be successfully discharged as well as that ALARP can be achieved for the whole project lifecycle:

- HI-2241088 Holtec Britain CDM Strategy [95]
- HI-2241089 SMR-300 GDA Holtec Nuclear Site Health and Safety Management System Report [96]

Furthermore, the construction of SMR-300 in the UK will benefit from the experience gained through the deployment of the SMR-300 at Palisades, near Covert, Michigan.

Traditionally, RC has been used for the construction of nuclear containment enclosure structures. The generic SMR-300 design utilises SC modular construction for the CES. Furthermore, the design and construction of other structures, such as the RAB and IB, may also evolve in the future to utilise modular construction to the extent possible.

SC construction provides the following benefits over RC:

- Eliminates the need for formwork and rebar assembly on site.
- Potentially provides better quality control since modules are fabricated in-shop.
- Potentially reduces construction overheads, since the modules are fabricated in-shop and on-site assembly may utilise automatic welding processes.

The overall outcome of the above is a potentially reduced construction programme, potentially greater quality control and potential health and safety benefits.

The design methodology and testing of the SMR-300 SC modular concept is in ongoing development. It will be available beyond the GDA. Holtec has been providing updates to ONR during the GDA through regular regulatory engagements.

The ongoing design development of the SMR-300 SC modular construction includes but is not limited to:

- Development of the design philosophy.
- Development of the test plan.
- Fabrication and confirmatory testing.
- Concrete mix design and casting.
- Development of module configurations.
- Supporting finite element modelling analyses for validation and benchmarking.
- Production of drawings, calculation packages and technical reports.
- Independent reviews of QA testing, analysis and documentation produced.
- Development of a Licensing Topical Report (LTR) for submission to the USNRC.

The basemat foundation of the CS and CES consists of a large RC raft foundation containing a significant quantity of steel reinforcement bars and embedments. A 3D 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.

It is recognised that Conformité Européenne (CE)/UK Conformity Assessed (UKCA) marking will be required for certain products (e.g., steel) that will be used for construction of the generic SMR-300 Civil Engineering SSCs. CE/UKCA marking and testing requirements to achieve CE/UKCA marking may influence the selection of products used for construction. Holtec has produced a Metrication Safety Justification Overview [67]. This sets out the strategy for metrication within the generic SMR-300, and how it will be demonstrated to be ALARP. To establish detailed procedures for achieving CE/UKCA marking, the extent of metrication of the generic SMR-300 will need to be established. The procedures that will be adopted in the design, manufacturing and construction process to ensure compliance with UK legislation in relation to product marking will be developed beyond GDA. Holtec has a track record in delivering nuclear safety related products in the UK and the European Union (EU) complying with legislative requirements.

20.7.4 CAE Summary

This sub-chapter presented the arguments and evidence associated with claim 2.2.11.4. The maturity of evidence provided is generally aligned with the expectations of a PSR.

QA and EIMT requirements for each nuclear safety related Civil Engineering SSC are defined in the respective design specification documents. The evidence outlined in this sub-chapter support the claim that the SMR-300 is designed taking into consideration EIMT requirements, so as to ensure that civil SSCs will continue to meet their design intent throughout the lifetime of the SMR-300, including provision for appropriate access, maintenance and inspection.

20.8 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the Civil Engineering chapter and how it contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5[37] sets out the overall approach for demonstration of ALARP and how contributions from individual chapters are consolidated.

This sub-chapter therefore consists of the following elements:

- Technical Summary.
- ALARP Summary.
    - Demonstration of Relevant RGP.
    - Evaluation of Risk and Demonstration Against Risk Targets.
    - Options Considered to Reduce Risk.
- GDA Commitments and Forward Actions.
- Conclusion.

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

20.8.1 Technical Summary

Part B Chapter 20 aims to demonstrate the following level 3 claim to a maturity appropriate for a PSR:

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 Civil Engineering SSCs have been designed using applicable US nuclear codes and standards. These have been compared against UK RGP and differences or potential risks in the application of the codes and standards in the UK context have been identified. These will be considered for implementation, taking into account considerations of the UK context and RGP.

Codes and standards have been developed based on OPEX from their use in the nuclear industry for over 70 years. OPEX from design, manufacture, construction and operation of nuclear safety related facilities have been used to develop and improve the design codes to ensure they reflect current learning and best practice.

The SMR-300 Civil Engineering SSCs are designed using best practice analysis and design methodologies which include:

- Design to applicable US nuclear codes and standards and UK RGP ensures that a reasonably conservative approach is adopted. Potential risks in the application of the adopted RGP codes and standards in the UK context are identified.
- A robust structural analysis and design methodology has been adopted which reflects UK RGP.

In addition, the safety and non-safety functions of civil SSCs have been defined and their classification determined taking into consideration the required functions during a DBE.

The SMR-300 Civil Engineering SSCs are designed using design and analysis methods which are considered RGP. 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.

QA and EIMT requirements for Civil Engineering SSCs are defined in the respective design specifications.

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.

20.8.2 ALARP Summary

20.8.2.1 Demonstration of RGP

The SMR-300 is designed in accordance with USNRC regulatory guidance and applicable CFRs. The standard design of the SMR-300 intends to comply with the USNRC requirements set forth in Title 10 CFR Part 50 [42].

Section 6.5 of ONR's TAG for Civil Engineering [64] Annex 1 identifies USNRC guidance as relevant good practice. NUREG-0800 (identified in Table 2 of this Chapter) is explicitly mentioned as an example.

HI-2241193 [97] presents a review of the ONR SAPs against the requirements of Title 10 CFR Part 50, Appendix A, GDC for nuclear power plants. The aim of this report is to:

- Demonstrate broad equivalency between the US SMR-300 safety principles and the equivalent thematic grouping of UK SAPs, to support the fundamental assessment of the SMR-300 design and to identify any potential gaps in equivalency.
- Provide confidence that the current principles against which the SMR-300 design is being developed will be met and where evidence for this will be presented.

HI-2240251 [66] presents the top-level plant design requirements of the SMR-300. The design requirements identified include Holtec SMR-300 objectives, EPRI requirements for nuclear power plants, as well as other relevant USNRC regulations and guidance. The EPRI requirements are aligned with WENRA and IAEA guidance.

The design of the SMR-300 Civil Engineering structures is aligned with RGP and USNRC requirements. The design adopts nuclear-specific codes and standards endorsed by the USNRC and internationally recognised bodies such as IAEA. The principal codes and standards identified within sub-chapter 20.4 are RGP. This is based on existing practices adopted on UK nuclear licensed sites, application in earlier and successful GDAs, and recognition by ONR TAGs.

For example, ASCE/SEI 4 [56] and ACI 349 [54] are indicated as RGP for seismic analysis of nuclear related civil engineering structures and blast/impact loading respectively in NS-TAST-GD-017 [64]. Several of the adopted principal codes and standards have no UK-equivalent codes and standards.

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 SC modular construction. The design of the SC modular concept is in development. It utilises experience, physical testing and numerical analysis, in alignment with ONR SAPs ECS.4 and ECS.5 [69].

The CES and CS structurally contain the AR. Furthermore, the overall design concept of the CS aims to reduce risks to ALARP by preferentially adopting passive engineered safety measures as a means to mitigate design basis accidents. This approach is judged to represent good practice. The CS design concept has been developed using a recognised industry standard, which is UK RGP. Exclusion of the exterior surface of the CS and the interior surface of the CES from the chemical environment of the AR is achieved through selection of coatings [33] [31]. Evaluation of the choice of coating for the CS [70] concludes that although the design standard adopted for the CS coating is proportionate to the classification of the CS and delivers the required reliability and assurance, the choice of coating is not currently well defined and a commitment has been raised to review the chemistry/coating arrangements and the ALARP position for the AR post-GDA. The structural analysis (seismic, aircraft impact, etc.) of the SMR-300 reactor building takes into account the coupled nature of the CES, AR and CS.

20.8.2.2 Lessons Learnt

To demonstrate understanding of ONR's expectations for the PSR, a review of the Regulatory Observations (RO) / Regulatory Issues (RI) relevant to Civil Engineering from previous GDAs [98], has been undertaken to ensure that RGP, OPEX and important lessons learnt from past GDAs are being considered.

A summary of the common lessons learnt related to Civil Engineering is provided below from ONR-GDA-GD-007 [99]:

1. *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*.
    - Classification and categorisation are addressed in sub-chapter 20.5. Demonstration of design basis analysis and cliff edge effects will be undertaken post-GDA when analysis results will be available. The effect of qualification of monitoring equipment and Class 1 barriers on civil engineering structures will be considered post-GDA.
2. *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. Refer to sub-chapter 20.5 and the Structural Analysis and Design Methodology Report [39].
3. *Extensive code comparison is required if bespoke codes are used in the design*.
   - The principal civil engineering codes and standards adopted for the SMR-300 are RGP. The applicability of these codes and standards in the UK context has been considered [41] [38].
4. *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 CES utilises a novel SC modular concept. A design methodology backed up by analysis and experimental testing is in development [68] in alignment with ONR SAPs ECS.4 and ECS.5 [69].
5. *The RP should consider that ONR has different assumptions regarding the type of aircraft involved in the impact load case*.
   - Aircraft impact hazards are considered in Part B Chapter 21 [21].
6. *The RP should consider the cliff edge effects from combined hazards*.
   - Beyond Design Basis Accident Analysis is considered in Part B Chapter 15 [14] and sub-chapter 20.6.3 of this Chapter. The combined hazards methodology for internal and external hazards is addressed in HI-2241054 [87].
7. *There are different assumptions on incident loads informed by internal hazards*.
   - Internal hazards are considered in Part B Chapter 22 [23].
8. *ONR requires the RP to understand (and mitigate) the risks associated with construction, commissioning, operations and decommissioning of the plant*.
   - Construction and commissioning are considered in Part B Chapter 25 [25]. Operations are considered in Part B Chapter 9 [10]. Decommissioning is considered in Part B Chapter 26 [26]. The management of Nuclear Site Health and Safety, which applies to the full lifecycle of all Civil Engineering SSCs, is considered in Part B Chapter 12 [17]. Refer also to sub-chapter 20.7.
9. *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 CS protected by a CES. The CS design pressure provides at least a [REDACTED] margin above the accepted peak calculated containment pressure following a Loss of Coolant Accident (LOCA) or main steam or main feedwater line break [33]
10. *Modern approaches to the design for fire safety of novel forms of construction are required*.
    - The design methodology and design of the novel SC modular concept is in ongoing development and will be available after the GDA. The design development will consider fire safety.

The design principles and current development of the generic SMR-300 provide confidence that these lessons, where relevant, can be addressed successfully.

20.8.2.3 Evaluation of Risk and Demonstration Against Risk Targets

The numerical targets against which the demonstration ALARP is considered can be found in Part A Chapter 2 [4]. Civil Engineering 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 (BSO) are considered broadly acceptable, and no further consideration of risk reduction is required. Risks between the BSOs and Basic Safety Levels (BSL) require a consideration of risk reduction options. Risks above the BSLs are not acceptable.

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, 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.

The evaluation of the normal operations and accident risks against Targets 1-9 is summarised in Part A Chapter 5 [37].

20.8.2.4 Options Considered to Reduce Risk

The process for the assessment of risk reduction options is presented in the Design Management process [73]. Part A Chapter 5 [37] considers the holistic risk-reduction process for the generic SMR-300.

A review of Design Challenges raised as part of the Design Management process [73] has been undertaken to assess whether any of the challenges have an impact on the Civil Engineering topic.

HI-2241524 Decision Paper on Containment Structure Design Change [72] introduced design changes to the general arrangement of the CES and CS for the US SMR-300 reference design for Palisades. Holtec Britain has provisionally assessed the impact of these prospective design changes in line with the Design Management process [73]. As part of the assessment, a Design Challenge was raised and HI-2250420 UK GDA Prospective Design Change - Containment Structure [74] was produced.

It was concluded that the prospective design changes do not have precedent for deployment in the UK and therefore will require further assessment that risks can be reduced to ALARP. However, given the operational experience of Holtec International and its partners (e.g., PARI), and the outcome of the optioneering performed by both Holtec International and Holtec Britain, it was recommended that these prospective design changes are endorsed for UK deployment, subject to the satisfactory completion of the further work indicated in the relevant prospective design change form [74].

20.8.2.5 GDA Commitments

GDA commitments which relate to this chapter have been formally captured in the Commitments, Assumptions and Requirements process [5]. Further details of this process are provided in Part A Chapter 4 [6]. The GDA commitments raised in Part B Chapter 20 are:

- Commitment C_Civi_012: The design methodology of the SMR-300 steel-concrete modular concept is developed using guidance from relevant nuclear-specific US codes and standards, coupled with physical testing and numerical analysis. This design methodology is currently in development and will not be available within the GDA timeframe. A Commitment is raised to demonstrate that the use of this design methodology enables the reduction of risks to ALARP. Target for Resolution - Issue of UK Pre-Construction SSEC.
- Commitment C_Civi_092: The Annular Reservoir is a design feature of the SMR-300 and offers considerable safety benefits to the plant. However, with the metal containment structure vessel surrounded with water in the Annular Reservoir and the exemption from post weld heat treatment (ASME Code Case N841), additional assessments are required on the containment structure to build a robust case for its continued integrity under fault load conditions. A Commitment is raised to conduct an assessment on the brittle fracture susceptibility of the containment structure including an assessment into the impact from thermal shock. Target for Resolution - Issue of Pre-Construction SSEC.
- Commitment C_Civi_093: The Annular Reservoir is a design feature of the SMR-300 and offers considerable safety benefits to the plant. Given the presence of water of the AR in contact with the containment structure, supplementary options assessments of degradation management solutions will be undertaken. A Commitment is raised to review the ALARP position of the specific AR degradation management arrangements and conduct further optioneering for options that were not explicitly considered in the US design basis. This Commitment does not include the water specification of the AR which has already been subject to a Design Decision. Target for resolution - Issue for UK Pre-Construction SSEC

20.8.3 Conclusion

This chapter summarises the Civil Engineering design for the SMR-300. It identifies the claims and arguments that will form the basis of the safety case for the Civil Engineering topic, throughout the lifecycle of SMR-300, to a maturity aligned to a PSR.

As the design and safety case are developed, evidence will be provided to substantiate these claims and arguments.

GDA Commitments have been raised.

It is therefore judged that the safety of the Civil Engineering design will be able to be demonstrated subject to resolution of the GDA commitments.

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20.10 LIST OF APPENDICES

Appendix A PSR Part B Chapter 20 CAE Route Map

A summary of the Safety, Security, and Environmental Case (SSEC) claims, arguments and evidence for the Civil Engineering topic is presented in Table 7.

Table 7: PSR Part B Chapter 20 CAE Route Map (Detailed)

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