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Holtec Britain Ltd	HI-2240341
Sponsoring Company	**Document Reference**
1	23 September 2025
Revision No.	**Issue Date**
Report	Non-proprietary
Record Type	**Proprietary Classification**
ISO 9001	No
Quality Class	**Export Control Applicability**

Record Title:

PSR Part B Chapter 10 Radiological Protection

Proprietary Classification

This record does not contain commercial or business sensitive information.

Export Control Status

Export Control restrictions do not apply to this record.

Revision Log

Revision	Description of Changes
0	First Issue to Regulators to support PSR R0
1	Second Issue to Regulators to support PSR R1

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10.1 INTRODUCTION

The Fundamental Purpose of the Generic Design Assessment (GDA) Safety, Security and Environment Case (SSEC) is to demonstrate that the generic Small Modular Reactor (SMR)-300 can be constructed, operated, and decommissioned on a generic site in the 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 Evidence (CAE) 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 10 of the PSR presents the CAE for the Radiological Protection topic.

10.1.1 Purpose and Scope

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

This chapter (Part B Chapter 10) links to the overarching claims through Claims 2.1 and 2.2:

Claim 2.1: The nuclear safety assessment specifies the requirements for safety measures such that safety functions are fulfilled, informs operational and emergency arrangements and demonstrates that risk is tolerable and ALARP.

Claim 2.2: The design of the systems and associated processes have been developed taking cognisance of Relevant Good Practice (RGP) and substantiated to achieve their safety and non-safety functional requirements.

As set out in Part A Chapter 3, Claims 2.1 and 2.2 are decomposed further. Claim 2.1 is further decomposed across several nuclear safety assessment disciplines which are responsible for the development of the nuclear safety assessment. Claim 2.2 is further decomposed across several engineering disciplines which are responsible for the development of the design of relevant Structures, Systems and Components (SSC).

In support of Claim 2.1, this chapter presents the Radiological Protection codes and standards that have been applied to the design of the SMR-300 and demonstrates how they are applicable to the operation of a reactor in the UK to ensure that doses to On Site Workers (OSW) and Members of the Public (MoP) are within UK dose limits and ALARP. This is decomposed into Claim 2.1.1.

In support of Claim 2.2 this chapter demonstrates, through Claim 2.2.15, that the design of SSCs with a Radiological Protection function considers RGP in meeting the Radiological Protection requirements to ensure that exposures to both OSW and MoP are minimised.

Claim 2.1.1: Radiological Protection requirements are identified such that doses to workers and public during normal operations are below legal limits, defined justified dose targets and constraints are ALARP.

Claim 2.2.15: Structures, Systems and Components (SSCs) are designed to meet Radiological Protection requirements and minimise exposures.

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 10.4.

This chapter covers the Radiological Protection aspects of normal operations, which include all normal operating states, including start-up, shut-down, and full-power operation, Examination, Inspection, Maintenance, and Testing (EIMT), refuelling, and abnormal operations. Outside of the reactor, normal operations include all radioactive source handling operations and fuel movements.

The radioactive sources included within the scope of this chapter are those within the following Nuclear Island (NI) structures as well as the Interim Spent Fuel Storage Installation (ISFSI):

- Containment Enclosure Structure (CES) including Containment Structure (CS) and associated systems.
- Reactor Auxiliary Building (RAB) and associated systems.
- Intermediate Building (IB).

The chapter is structured as follows:

- Sub-chapter 10.2 introduces the source terms used in the SSEC.
- Sub-chapter 10.3 provides an overview of Radiological Protection for the SMR-300.
- Sub-chapter 10.4 details the CAE relevant to Radiological Protection that will be addressed within this chapter.
- Sub-chapter 10.5 covers the Radiological Protection requirements and associated arguments and evidence.
- Sub-chapter 10.6 provides the arguments and evidence regarding dose management.
- Sub-chapter 10.7 provides the arguments and evidence that the implementation of Hierarchy of Controls (HoC) makes risks ALARP.
- Sub-chapter 10.8 provides the arguments and evidence that sources are appropriately characterised.
- Sub-chapter 10.9 provides the arguments and evidence that shielding is designed to minimise exposures.
- Sub-chapter 10.10 provides the arguments and evidence that material selection minimises neutron activation,
- Sub-chapter 10.11 provides arguments and evidence for the ventilation and containment systems design.
- Sub-chapter 10.12 provides arguments and evidence that monitoring, and alarm systems reduce doses and risks to ALARP.
- Sub-chapter 10.13 provides the arguments and evidence for the designation of areas.

- Sub-chapter 10.14 provides preliminary arguments and evidence for out-of-core criticality safety in the SMR-300 design.

Finally, sub-chapter 10.15 provides a technical summary of how the claims for this chapter have been achieved, together with a summary of key contributions from this chapter to overall ALARP. Sub-chapter 10.15 also discusses any GDA commitments that have arisen.

Excluded from the Part B Chapter 10 scope are:

- Radiological impact on MoP and the environment from all discharge routes, which is covered in Preliminary Environmental Report (PER) Chapter 3 Radiological Impact Assessment (RIA) [3] and radioactive waste transfers, see Part B Chapter 13 Radioactive Waste Management [4].
- Non-radiological environmental impact from normal operations, which is covered in PER Chapter 4 Conventional Impact Assessment [5].
- Radiological consequence analysis for fault conditions and the minimisation of accident source terms. These are addressed in Part B Chapter 14 Design Basis Analysis (Fault Studies) [6].
- Decommissioning and the provision of information, Radiological Protection and the design for decommissioning are covered in Part B Chapter 26 Decommissioning Approach [7].

A master list of definitions and abbreviations relevant to all PSR chapters can be found in Part A Chapter 2 General Design Aspects and Site Characteristics [8].

10.1.2 Assumptions

There are no assumptions raised with respect to Part B Chapter 10.

10.1.3 Interfaces with Other SSEC Chapters

The Radiological Protection chapter interfaces with the following PSR and PER chapters.

Part A Chapter 2 [8] presents an overview of the design evolution, the generic design of the SMR-300 and a description of the generic site envelope presented for the GDA, the fundamental design and safety principles and the reference design for the generic SMR-300. It also covers and gives a detailed account of the layout/configuration process and design of the Nuclear Steam Supply System (NSSS) and plant as driven by the design philosophy and top-level plant requirements. Part A, Chapter 2 also introduces the Numerical Targets (NT) for radiation exposure. Further details of the public exposure groups and the related assumptions regarding the generic site description are provided in PER Chapter 3 [3].

Part A Chapter 1 [1], Part B Chapter 2 Reactor [9] and Part B Chapter 5 Reactor Supporting Facilities [10] provide a description of the SSCs within the NI of the SMR-300. This includes radiation sources including the fuel and core, spent fuel storage (in-containment), primary and secondary coolant systems and ventilation systems. Part B Chapter 2 and supporting document GDA Fuel Design Criteria and Limits [11] describe the limits to which the plant will be designed and operated in order to prevent radioactive release.

Part B Chapter 4 Control and Instrumentation Systems (I&C) [12] describes the key functions of I&C systems, their design principles, and their significance in maintaining safe and reliable

operation. Radiological protection SSCs (e.g., instrumentation for radiation monitoring) that are I&C-based will be substantiated within the I&C topic.

Part B Chapter 11 Environmental Protection [13] and Part B Chapter 13 [4] describe the Best Available Techniques (BAT) and optimisation of the generation of Low Level Waste (LLW) and Intermediate Level Waste (ILW) that contributes to the demonstration that effective doses are ALARP. PART B Chapter 13 [4] also describes the design of the radwaste systems, effluent discharge systems e.g., filter / resin changes to be optimised with worker dose and discharges. PER Chapter 6 Demonstration of Best Available Techniques [14] describes the approach to and application of BAT (and therefore the UK definition of the As Low As Reasonably Achievable (ALARA) principle as applies to all aspects of the management of radioactive substances and wastes, including their disposal) for the SMR-300. These chapters demonstrate how BAT works alongside ALARP to provide a holistic optimised solution.

PER Chapter 3 [3] assesses the dose to MoP from radioactive effluent discharges and direct radiation. The Radiological Protection topic provides the direct radiation dose rate data. Assessed total doses to MoP are compared against dose limits in the Ionising Radiations Regulations 2017 (IRR17) [15] and constraints in The Environmental Permitting (England and Wales) Regulations 2016 (EPR16) [16].

Part B Chapter 20 Civil Engineering [17] describes the building containment and structures of the NI, including the CES, CS, RAB and ISFSI. Shielding is designed to provide protection to potential exposed groups.

PART B Chapter 14 [6] presents the deterministic design basis analysis for the SMR-300 following accident conditions and presents the basis for demonstration that the risk is ALARP in comparison with the NTs introduced in Part A Chapter 2 [8]. Safety functions for SSCs (including Radiological Protection) following design basis accidents are derived here. This will include those internal and external hazards identified in Holtec SMR GDA Part B Chapter 21 External Hazards [18] and Holtec SMR GDA Part B Chapter 22 Internal Hazards [19].

Holtec SMR GDA Part B Chapter 15 Beyond Design Basis, Severe Accidents Analysis and Emergency Preparedness [20] provides an assessment of the SMR-300 following accident conditions that are low in frequency or high in consequence and derives safety functions for SSCs (including Radiological Protection) under these accident conditions. Emergency planning and preparedness requirements are also addressed here, notably the radiological exposures where emergency measures will apply.

Part B Chapter 16 Probabilistic Safety Assessment [21] presents the probabilistic analysis for the SMR-300 following accident conditions and derives safety functions for SSCs (including Radiological Protection) to meet probabilistic risk criteria.

Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [22], Part B Chapter 9 Conduct of Operations [23], Part B Chapter 25 Construction and Commissioning [24], Part B Chapter 12 Nuclear Site Health and Safety and Conventional Fire [25], Part B Chapter 17 Human Factors [26], and Part B Chapter 26 Decommissioning Approach [7] provide a description of the processes and procedures that will result in the management of risks during the respective lifecycle phases. Part B Chapter 9 [23] also include the process for development of EIMT schedules and the description of reactor operating states. Part B Chapter 17 [26] describes the improvements made from a review of Operating Experience (OPEX). Part B

Chapter 26 [7] considers source terms for the decommissioning assessments, the design principles and provisions that facilitate decommissioning, and the main Radiological Protection considerations during decommissioning.

Part B Chapter 23 Reactor Chemistry [27] describes the normal operating source term of the Reactor Coolant System (RCS) and demonstrates that this is minimised by the chemistry regime, material selection, and controlling the accumulation of radionuclides. Additionally, Reactor Chemistry demonstrates that the calculated normal operational source term is representative of the SMR-300 plant and appropriate for the design stage.

Part B Chapter 24 Fuel Transport and Storage [28] describes the movement, handling and storage of new and used fuel outside of containment.

Part A Chapter 5 Summary of ALARP [29] is intended as a summary of the SSEC and naturally links to all SSEC reports and their respective chapters.

Part B Chapter 19 Mechanical Engineering [30] describes how the design of the systems and associated processes have been developed to achieve their safety and non-safety functional requirements. Also, the overall design and architecture of mechanical SSCs ensure that safety functions and non-safety functions are delivered, and faults arising from failures of the SSCs are minimised.

10.2 SOURCE TERMS

Source terms are critical in describing the type and quantity of radionuclides in the reactor core and related systems to determine the magnitude of radiological hazard, inform component design and enable assessment of risk to OSW, MoP and the Environment.

Source terms quantify the radiological inventory of the SMR-300 originating from the mix of radionuclides released from the fuel. The source term is expressed in terms of fractions of fission product inventory in the fuel, their physical and chemical form, the timing of their release and the activation of materials present in the plant from the neutron flux emanating from the reactor core. Determining source terms is a critical part of system design to understand the nature of radiation hazard in the core that extends to other parts of the system. Source terms are used to ensure SSCs are designed and developed to provide adequate protection throughout the various parts of the plant. This is key to ensuring doses to OSW and MoP are within limits and ALARP, and resulting impacts from Radioactive Waste are ALARA, through application of BAT and the Waste Hierarchy.

Development of the source terms shall:

- Cover all significant radionuclides.
- Cover all systems which are expected to contain radioactivity.
- Cover all operational states.
- Cover all appropriate sources of radioactivity within the plant, including mobile and fixed sources.
- Consider how the nature and quantity of radioactivity within the plant may change over time.
- Be consistent with how the defined source terms are used by, and support, the safety and environmental case for SMR-300.
- Be consistent with the design and operations of SMR-300.

Justification of source terms shall provide robust supporting evidence for the defined source term, demonstrated to be appropriate for the SMR-300 and consistent with the extant safety and environmental cases where relevant. This is substantiated in Evaluation of SMR-300 Calculated Source Terms Against Publicly Available Information [31] which compares the calculated source terms of the RCS for the SMR-300 generic design to publicly available data measured at operational nuclear plants and also compares to other estimations from similar stages of design development. This document focuses on the RCS only; further work is required to substantiate the remainder of the source terms defined for the SMR-300.

So far, the developed source terms for SMR-300 consist of a realistic primary source term and Design Basis source term for each developed inventory. The realistic source terms are best estimates of the source term values over a representative condition for Normal Operations. The realistic source term is used in calculations for radioactive waste inventories, disposability assessments and routine radiological discharges. Design Basis source terms are considered bounding limits for plant design therefore, they are important for key safety-related applications such as derivation of shielding requirements and radiological consequences of fault and accident assessments to ensure that doses to the OSW and MoP are minimised.

Given their different generation mechanisms and subsequent behaviour in the plant, the definition and justification approach for each nuclide category differs. The source terms are

split into various nuclide categories, based on their generation mechanism; namely Noble Gases, Halogens, Rubidium and Caesium, Other Fission Products, Activated Corrosion Products, and Water Activation Products. An overview of the various nuclide categories, the methodologies used to derive the fission product, coolant activation product and corrosion product source terms and rationale are provided in the SMR-300 Calculated Source Terms Evaluation [31]. The entire derivation of the source term for the different radionuclides for the RCS source term is provided in Contained Radiation Sources for Normal Operation [32].

10.2.1 Source Terms used in the SSEC

The source terms are an essential reference for many of the PSR and PER chapters but in particular (specifically but not exclusively) those presented in Table 1.

Table 1: Application of Source Term by SSEC Chapter

Chapter	Source Term Application
B10 Radiological Protection	Shielding calculations and determining radiological inventories to enable dose assessments
B13 Radioactive Waste Management [4] and PER 1 Radioactive Waste Management Arrangements [33]	Used to classify waste, develop radioactive waste inventories, and recommend waste management arrangements.
B14 Safety and Design Basis Accident Analysis [6]	Determination for mitigation against the loss of SSCs to ensure safety of OSW, MoP and the environment. The normal condition core inventory was used to inform the development of the PFS.
B15 Beyond Design Basis, Severe Accidents Analysis and Emergency Preparedness [20]	Contingency planning for the potential of off-site releases.
B23 Chemistry [27]	Demonstrate the impact of material choices, operating chemistry and operating practices on radioactivity in the plant and to show that these reduce radioactivity So Far As Is Reasonably Practicable (SFAIRP).
B26 Decommissioning Approach [7]	Characterise items and prepare decommissioning waste inventories.
PER 2 Quantification of Effluent Discharges and Limits [34]	Used to determine the effluent inventory for gaseous and aqueous radioactive discharges.
PER 3 Radiological Impact Assessment [3]	Applies the discharge limits set in PER 2 in assessment of the radiological impact of effluent discharges to MoP and the environment.
PER 5 Approach to Monitoring and Sampling [35]	Used in support of selection of appropriate methods for environmental sampling and monitoring.
PER 6 Demonstration of Best Available Techniques [14]	Used in support of assessments to identify BAT across waste generating and management systems.

10.2.2 SMR-300 Source Terms and Methodology Summary

Source terms for SMR-300 will continue to be developed and modified as the design develops; however, the current source terms that have been developed for SMR-300 are listed in the table below. The source terms available reflect the current maturity of the design; these are however, considered sufficient to complete a fundamental Step 2 GDA assessment. Source terms are presented in Table 2, these consist of those that have been developed and published together with source terms still under development within Step 2.

Table 2: SMR-300 Source Terms

Source Term	Sources Covered	Sub-Chapter	Application
SMR-300 Source Terms [36].	Core Isotopic Inventory, Total Core Decay Heat	10.2.2.1	Shielding assessments, worker dose assessments, public dose assessment (direct shine only), DBAs etc, Fault Studies.
SMR-300 Contained Radiation Source for Normal Operations [32].	Primary and Secondary Source Terms (Realistic and Design Basis)	10.2.2.2 and 10.2.2.3	Shielding Assessments, Worker Dose Assessments, Public Radiological Discharge Calculations, Radioactive Waste Inventories, Radioactive waste system design, disposability assessments, Decommissioning activities.
SMR-300 Spend Fuel Pool Cooling System (SFC) Source Terms (in progress) [37].	Spent Fuel Pool (SFP) coolant during different modes of operation, SFC system equipment	10.2.2.4	Shielding assessments, worker dose assessments, radioactive waste Inventories, Radioactive waste system design, disposability assessments, decommissioning activities, airborne activity calculations in containment atmosphere to enable occupational dose exposure and to support the development of the aqueous and gaseous effluent inventory in PER Chapter 2 [34]
Estimate of the SMR-300 Gaseous and Liquid Effluent Releases using the Gaseous and Liquid Effluents - Pressurised Water Reactors (GALE-PWR) 3.2 Code [38]	Liquid Radiological Waste (LRW) and Gaseous Radiological Waste (GRW)	10.2.2.5.2	
Calculation of SMR-300 Solid Radiological Waste Inventories [39]	Solid Radiological Waste (SRW)	10.2.2.5.1	
Airborne Source Terms (in progress)	RAB (to be confirmed)	10.2.2.5.3	
Secondary Source terms (in progress)	Activated in core instrumentation, activated	To be determined	Shielding assessments, worker dose assessments

10.2.2.1 Reactor Core Source Term

The fuel assembly and core inventory source term is the source term from which all other source terms are generated. Neutron and gamma source terms (including decay heat values and quantities of radionuclides) are calculated with the TRITON and ORIGAMI/ORIGEN sequences of the SCALE 6.2.1 code package. TRITON generates the fuel assembly cross-section libraries. ORIGAMI/ORIGEN performs nuclear fuel depletion for a given axial power profile, burnup, and enrichment combination and produces the resulting source terms at desired cooling times. The neutron source term is comprised of alpha-n reactions, spontaneous fissions, and delayed neutrons. The gamma source term is comprised of fission products and actinides. Detailed methodology is documented in SMR-300 Source Terms [36].

10.2.2.2 Primary Coolant

The principal mechanisms of production of radioactive materials in the primary coolant under reactor normal operating conditions are:

- Fission product leakage to the coolant from defects in the fuel cladding and fission product generation in tramp uranium.
- Activation of the corrosion products in the reactor core.
- Activation of the coolant water and its chemical additives in the reactor vessel.

Methodology used for the calculation of the Primary Source Term is using American National Standards Institute(ANSI)/American Nuclear Society (ANS), Radioactive source term for normal operation of light water reactors ANSI/ANS-18.1 (2016) [40] and the calculations and

results for both the Realistic and Design Basis source terms are documented in Contained Radiation Sources for Normal Operation [32].

10.2.2.3 Secondary Coolant

Leakage of primary coolant into the secondary coolant via steam generator tubing results in minimal levels of radioactivity in the secondary coolant system. Water or steam leakage from the secondary system inputs to the liquid and gaseous radwaste treatment systems. The methodology, calculations and results for both the Realistic and Design Basis source terms are documented in Contained Radiation Sources for Normal Operation [32].

10.2.2.4 Spent Fuel Pool Coolant

The SFP coolant source term considers the activity of the SFP coolant during different modes of plant operation. It is considered that radionuclides occur in SFP coolant due to mixing with reactor coolant and spallation of fuel rod crud during refuelling process. The radionuclides enter the Spent Fuel Pool Cooling System (SFC) with refuelling water transferred from the Refuelling Water Storage Tank (RWST) after mixing with RCS during cooldown and due to crud spallation from spent fuel rods. Activity of crud is considered as the main contributor to SFC activity.

Specific activities of the SFP coolant due to mixing with RCS at the beginning of the refuelling mode is determined based on calculations of the water mass balance between RCS, RWST, Passive Core Makeup Water Tank (PCMWT) and SFC during reactor cooldown and refuelling process provided in System Design Description (SDD) for the Spent Fuel Pool Cooling System [41]. Detailed methodology is documented in SMR-300 Spent Fuel Pool Cooling System Source Terms Report [37]. Results of this analysis are used as inputs for the calculation of the source terms of SFC components, airborne radionuclide concentrations in containment atmosphere and occupational dose estimations etc.

10.2.2.5 Calculated Waste Source Terms

The following source terms have been determined in support of the GDA to provide calculations to indicate activities for the purposes of radioactive waste inventories, gaseous and aqueous-liquid radioactive waste discharges and airborne activity concentrations.

10.2.2.5.1 Solid Radwaste

Preliminary estimation of the activity concentration in filter media in the Chemical and Volume Control system (CVC), SRW and LRW to support development of the SRW design and PER Chapter 1 Radioactive Waste Management Arrangements [33] The methodology for calculation the results of the SRW inventory is documented in Calculation of SMR-300 Solid Radiological Waste Inventories [39]. This source term will be utilised to develop shielding design for CVC, SRW and LRW plant items, for planning the designation of areas, and for the development of dose assessment to OSW and MoP from these systems and Radioactive Waste Storage and Treatment facilities.

10.2.2.5.2 Liquid and Gaseous Radwaste

Preliminary estimation of gaseous and liquid effluent releases from the SMR-300 in support of estimation of discharge limits and assessment of radiological impact on OSW, MoP and the environment for the GDA. The gaseous and liquid effluent releases are determined for a single

SMR-300 using the methodology from the US Nuclear Regulatory Commission (NRC) GALE-PWR code Version 3.2 that implements the methodology detailed in NUREG-0017 [42]. Detailed methodology and calculation of results are documented in Estimate of the SMR-300 Gaseous and Liquid Effluent Releases [38]. SRW source terms will also be utilised to determine shielding, radiation and contamination zoning and dose assessments.

10.2.2.5.3 Airborne Concentrations

Preliminary estimates of the concentrations of airborne radioactive material in the SMR-300 RAB and containment have been generated in support of the definition of the gaseous effluent discharge limits for GDA; however, these estimates are still under development. Maximum, steady-state concentrations of radioactive material are calculated for normal and refuelling conditions as well as Anticipated Operational Occurrences (AOOs) in both containment and the RAB. Inputs and assumptions are based on preliminary design documents that were used to develop the estimate of airborne concentrations.

10.3 OVERVIEW OF RADIOLOGICAL PROTECTION

10.3.1 Introduction

Radiological protection involves the protection of OSW and MoP from the effects of ionising radiation caused by man-made and natural sources of radiation during normal operations and fault conditions. In the context of SMR-300 this includes radiation from reactor operations and treatment, storage and discharges to the environment of radioactive waste. It also includes the prevention and minimisation of radiation risk as a result of out-of-core criticality accidents.

10.3.2 Radiological Protection Step 2 Deliverables

The Radiological Protection deliverables produced during Step 2 GDA are summarised in Table 3.

Table 3: Step 2 Deliverables

Title	Scope
UK / US Radiological Protection Regulatory Framework Comparison [43]	Provides a summary of the UK and US legislative framework for Radiological Protection in nuclear power generation, compares the two, and highlights any differences that need to be taken into consideration for the generic SMR-300 design.
Holtec SMR-300 Design Standard for Radiation Protection [44]	Describes the Radiological Protection philosophy and design requirements to OSW, MoP, and the environment against the radiological hazards.
SMR-300 IRR17 Compliance Matrix [45]	Provides a comparison of the SMR-300 design against the requirements of Ionising Radiation Regulations 2017 (IRR17) and highlights any areas of non-compliance.
SMR-300 Dose Management Strategy [46]	Provides the methodology for assessing the dose to workers as the design develops to support design development. Provides the rationale behind the dose constraints presented in the Radiation Protection Design Standard (RPDS). Provides discussion on how the RPDS will be implemented. Specifies the strategy for dose optimisation.
Holtec SMR-300 Shielding Basis of Design [47]	Provides the key assumptions, basic data and assessment criteria for shielding calculations. Provide preliminary shielding calculations.

10.3.3 Application of Radiological Protection across the SSEC

Radiological Protection is applicable across all parts of the SSEC including:

- Nuclear Safety.
- Environmental Protection.
- Security.
- Nuclear Material Safeguards.

As discussed in sub-chapter 10.1.3, Radiation Protection interfaces with many topic areas across the PSR and is fundamental to nuclear safety.

Environmental Protection encompasses protection of MoP and the environment. This is addressed in both IRR17, for sources of direct radiation, and EPR16 [16] for radioactive discharges, as discussed in sub-chapter 10.1.3.

The interface between Radiation Protection and Security is largely that of dose optimisation (application of ALARP) when performing security functions.

The Office for Nuclear Regulation (ONR) Nuclear Material Accountancy, Control and Safeguards Assessment Principles (ONMACS) [48] discuss the need for material controls via Fundamental Safeguards Expectations (FSE) 6 to 10 into the design of systems for the purposes of Nuclear Material Accountancy and Control Systems (NMACS). These align with regulation 29 of IRR17 – Accounting for Radioactive Substances.

10.3.4 Metrication

A systematic review of metrication issues has been undertaken across all disciplines. The primary metrication issue in Radiological Protection is related to dosimetric units. The system of units used in the design is predominantly imperial, and units for radioactivity and dose differ from the SI units generally used across the rest of the world, including the UK. To maintain consistency within this document, all measures will be presented in units consistent with UK regulatory expectations. Conversions for each unit taken from the US source material are presented in Table 4.

Table 4: Metrication - Conversion between US and UK units

Measure	US Unit	UK Unit	Conversion Factor
Radioactivity	Curie (Ci)	Becquerel (Bq)	1 Ci = 3.7×10^{10} Bq
Activity concentration (area)	Disintegrations per minute per 100 cm2 (dpm/100 cm^2)	Bq m2	1 dpm/100 cm^2 = 16.67 Bq m^{-2}
Activity concentration (volume)	pCi per litre (pCi L-1)	Bq m3	1 pCi L-1 = 0.037 Bq m^{-3}
Specific Activity	Ci g-1	Bq kg-1	1 Ci g-1 = 3.7×10^{10} Bq kg-1
Absorbed Dose	Rad	Gray (Gy)	1 rad = 0.01 Gy
Dose Equivalent	Rem	Sievert (Sv)	1 rem = 0.01 Sv
Exposure	Roentgen (R)	Coulomb/kilogram (C kg-1)	1 R = 0.000258 C kg-1

10.3.5 Description of Radiological Protection SSCs

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

Chapter A2 of the PSR [8] gives the site layout and a description of the main buildings. This chapter refers to the following SSCs, which have Radiological Protection safety functions:

- CVC System.
- Primary Sampling System (PSL).
- Containment Building Ventilation System (CBV).
- Control Room Normal Ventilation System (CRV).
- Radiologically Controlled Area (RCA) Heating, Ventilation and Air Conditioning (HVAC) system (RCV).
- GRW System.
- LRW System.
- SRW System.
- Radiation Monitoring System (RMS).
- CES.
- CS.
- RAB.

Detailed information on the above SSCs is provided in Part B Chapter 1 Reactor Coolant Systems and Engineered Safety Features [49] and Part B Chapter 5 [10]. A high-level description of these SSCs can be found in the sub-chapters below. Safety functions for Radiological Protection SSCs are listed in Appendix C. Commitment C_Faul_103 has been made in Chapter B14 to complete a comprehensive safety analysis for all systems, which will include identification of any further Radiological Protection safety functions, and classification of any further Radiological Protection SSCs.

10.3.5.1 CVC

The CVC [50] controls the water chemistry of the reactor coolant system inventory to ensure radiological source terms in the RCS are minimised during operation among other functions. The CVC includes demineralisers and filters that remove radioactive species from RCS coolant ensuring these are sequestered in preferential locations allowing for controlled transfer to SRW and LRW. The CVC also allows for controlled removal and transfer of activated RCS fluid to the GRW.

10.3.5.2 PSL

The PSL [51] delivers liquid and gaseous samples from various points in containment and the RAB to a centralised location. The primary sampling panel is designed to permit sampling during all modes of plant operation, including power generation, shutdown, refuelling, startup, and post-accident conditions without requiring access to containment. This ensures exposure to plant personnel during sampling are ALARP and chemical and radiological conditions of the Reactor Coolant (RC) are kept under constant review. Local grab sampling points are included on some systems which allow for local collection of samples. This reduces radiological exposure by reducing the amount of pipework containing radioactive volumes of fluid.

10.3.5.3 CBV

The CBV [52] is designed to control the containment temperature within a suitable range, prevent damage to the containment structure or equipment inside containment, and allow conditions for personnel to perform work, when accessible. The system is designed to control airborne contamination to within acceptable limits. The CBV forms the containment boundary in conjunction with doors and seals. The containment purge system is also used to purge the containment to reduce airborne contamination levels inside containment before to entry during outages.

10.3.5.4 CRV

The CRV System is designed to provide a reliable source of heating, ventilation, and cooling to areas served when Alternating Current (AC) power is available. Redundant safety-related radiation monitoring sample line connections are located upstream of the outside air intake isolation dampers. These monitors initiate operation of the non-safety-related supplemental air filtration unit on high airborne radioactivity concentrations and isolate the Control Room Emergency Zone (CREZ) on high-high particulate or iodine radioactivity concentrations.

10.3.5.5 RCV

Filtration of incoming and exhaust gases ensures habitability and minimisation of radioactive exposure to plant personnel. The generic SMR-300 includes a number of ventilation systems designed to control and minimise the airborne and gaseous radiological contamination

sources present in the operation of the plant. These ventilation systems include the RCV and CBV. Protection from airborne radioactive material is achieved through maintaining air pressure gradients and airflows from areas of low potential airborne contamination to areas of higher potential contamination to limit the spread of airborne contamination within the plant.

10.3.5.6 GRW

Although the GRW [53] predominantly provides protection by processing and minimising radioactive gaseous effluents discharged off-site, it also performs a function of protecting plant personnel on site by removing gaseous wastes in the RCS via the CVC. This ensures fission and activation products generated in the core, which can increase radiological dose, are removed and processed via storage and filtration prior to off-site release, ensuring doses to MoP from discharges are kept ALARA. 2.1.1. Particulate contaminants are filtered via High Efficiency Particulate Air (HEPA) filters in the HVAC system before discharge.

10.3.5.7 LRW

The LRW [54] is designed to protect plant personnel from radiation exposure, ensure doses are ALARP and meet the BAT requirements for radionuclide release to the environment. The LRW collects and processes wastes produced in the plant during normal operation and AOOs, including shutdown, refuelling, and maintenance. The LRW consists of tanks, pumps, filters and demineralisers that process radioactively contaminated wastes.

10.3.5.8 SRW

The SRW [55] collects, processes, packages, and stores radioactive solid wastes generated from normal plant operations, including AOOs. These wastes include spent resins, filter bed media, filter cartridges, HVAC filters, tools, Personal Protective Equipment (PPE), and other potentially contaminated wastes. The SRW consists of three major waste streams: wet solid wastes, dry solid wastes, and miscellaneous wastes. Design features are incorporated to maintain exposure to plant operation and maintenance personnel ALARP. These features include remote system operation, line flushing, and shielding of components and piping containing radioactive materials.

10.3.5.9 RMS

The RMS [56] is designed to perform plant effluent monitoring, process fluid monitoring, airborne radioactivity monitoring, and continuous indication of the radiation environment in plant areas where such information is needed.

10.3.5.10 CES

The CES [57]. is a cylindrical protective structure that fully envelops the CS. The CES is designed to provide shielding to the plant and personnel from radioactive sources inside the CS during power operations and postulated accidents.

The CES is constructed using steel-concrete (SC) modular walls for the above and below grade portions. It shares a common reinforced concrete base mat 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 (CLSM). The Annular Reservoir (AR) is formed between the walls of the CES and CS.

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 the replacement of major components and for access during a refueling or maintenance outage.

10.3.5.11 CS

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 reinforced concrete base mat shared with the CES. Above grade, its outer side interfaces with the AR, whereas below grade, its outer side interfaces with the CLSM fill.

The CS is designed to 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 [58].

10.3.5.12 RAB

The RAB is a five-storey building, with three storeys located above grade level and two storeys located below grade level. It houses safety-related and non-safety related SSCs. The RAB is designed to be constructed mostly from RC, although its construction may evolve to use 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 Equipment Hatch and the Personnel Hatch. The RAB also contains the Plant Vent Stack. Potentially contaminated effluents would be directed to the plant vent for monitoring before being released into the atmosphere. CBV effluent would also be routed to the Plant Vent Stack.

The RAB is designed to provide Radiological 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.

Further details on the RAB can be found in Design Specification for Reactor Auxiliary Building [59].

10.4 RADIOLOGICAL PROTECTION CLAIMS, ARGUMENTS AND EVIDENCE

This chapter presents the Radiological Protection aspects for the generic SMR-300 and therefore directly supports Claim 2.1.1 and Claim 2.2.15.

Claim 2.1.1: Radiological Protection requirements are identified such that effective doses to workers and public during normal operations are below legal limits, defined justified dose targets and constraints are ALARP.

Claim 2.1.1 has been further decomposed within Part B Chapter 10 into three Level 4 claims. These claims cover UK Radiological Protection legislative requirements, paying particular attention to dose limitation and hierarchy of controls. They also aim to address the limitation and optimisation Radiological Protection principles.

Table 5 shows the breakdown of Claim 2.1.1 and identifies in which section of this chapter these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 5: Breakdown of Claim 2.1.1 and where sub-claims are addressed

Claim No.	Claim	Chapter Section
2.1.1.1	The Generic Holtec SMR-300 is compliant with relevant Radiological Protection legislative requirements.	10.5 Radiological Protection Requirements
2.1.1.2	Doses associated with the generic SMR-300 are below legal limits and are demonstrated to be ALARP.	10.6 Managing Doses
2.1.1.3	The safety hierarchy of controls will be applied to ensure radiological hazards will be reduced to ALARP.	10.7 Hierarchy of Control

Claim 2.2.15: SSCs are designed to meet Radiological Protection Requirements and minimise exposures.

Claim 2.2.15 has been further decomposed within Part B Chapter 10 into seven Level 4 claims covering a number of Radiological Protection related SSCs. This claim also covers layout and zoning as these are related to shielding, ventilation and containment SSCs.

Table 6 shows the breakdown of Claim 2.2.15 and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 6: Breakdown of Claim 2.2.15 and where sub-claims are addressed

Claim No.	Claim	Chapter Section
2.2.15.1	Sources of residual radiological risk are appropriately characterised to inform the design of SSCs.	10.8 Sources are Appropriately Characterised
2.2.15.2	The generic Holtec SMR-300 shielding is designed and substantiated to minimise exposures for all plant areas and operation stages, including waste package transport.	10.9 Shielding is Designed to Minimise Exposures
2.2.15.3	Materials selection minimises the generation of neutron activation products in SSCs.	10.10 Material Selection
2.2.15.4	Ventilation and containment systems are designed to meet Radiological Protection requirements and minimise exposures.	10.11 Ventilation and Containment Design

Claim No.	Claim	Chapter Section
2.2.15.5	The design and layout of the generic Holtec SMR-300 includes monitoring and alarm systems to alert to the presence of radiation and contamination.	10.12 Monitoring and Alarm System
2.2.15.6	Radiation zoning ensures access to and between Controlled and Supervised Areas will be suitably controlled to minimise exposures to ALARP and prevent the spread of radioactive material.	10.13 Designation of Areas
2.2.15.7	Criticality controls are implemented to ensure that criticality risks are reduced to ALARP.	10.14 Criticality Safety

Appendix D provides a full CAE mapping for this chapter, which includes all lower-level CAE 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.

10.5 RADIOLOGICAL PROTECTION REQUIREMENTS

Claim 2.1.1.1: The generic Holtec SMR-300 is compliant with relevant Radiological Protection legislative requirements.

Claim 2.1.1.1 has been further decomposed into five arguments to address the claim.

- Relevant UK legislation has been mapped against US legislation to ensure all UK requirements are met or exceeded (A1).
- Significant gaps which would affect the design and operation of the SMR-300 have been raised as risks (A2).
- Resolutions to regulatory gaps have been identified and actioned (A3).
- ONR Safety Assessment Principles (SAPs) Radiological Protection Principles (RP) RP.1-RP.7 will be utilised in the development of design and procedures and in the assessment of the design to demonstrate the application of ALARP (A4).
- Documentation and assessments will be developed following the Approved Code of Practice (ACoP) to IRR17 (A5).

Argument 2.1.1.1 – A1: A mapping exercise of relevant UK legislation against US legislation has been undertaken to ensure all UK requirements are met or exceeded.

10.5.1 Evidence for Claim 2.1.1.1 – A1

UK-US Radiological Protection Regulatory Framework Comparison [43]

A comparison between UK and US legislation relating to Radiological Protection, as well as a detailed assessment of the design of the SMR-300 against UK legislation and guidance has been conducted and is reported in the UK and US Legislative Framework Comparison.

SMR-300 IRR17 Compliance Matrix [45]

The generic SMR-300 design has been assessed against the legislative requirements in IRR17 [15] and associated ACoP [60]; ONR SAP [61] principles RP.1-RP.7, NT.1-NT.3 and Engineering Principle- Criticality Safety (ECR) ECR.1 and ECR.2; relevant ONR Technical Assessment Guides (TAG), and guidance in IAEA Safety Standard: Radiation Aspects of Design for Nuclear Power Plants, Specific Safety Guide, SSG-90 [62] within the SMR-300 Compliance Matrix.

10.5.2 Narrative for Claim 2.1.1.1 – A1

The generic SMR-300 is designed in accordance with US NRC Code of Federal Regulations (CFRs) for nuclear power and associated regulatory guidance. The generic design of the SMR-300 was developed in compliance with US NRC requirements set forth in US NRC 10 CFR-20 Standard for Protection Against Radiation [63] and US NRC 10 CFR-50 Domestic Licensing of Protection and Utilisation Facilities [64]. Several guides and standards, summarised in Table 7, are also used to inform the design of the generic SMR-300.

Table 7: SMR-300 Codes and Standards

Label	Title	Revision
U.S. NRC Title 10, Code of Federal Regulations (CFR) 50	Domestic Licensing of Production and Utilisation Facilities [64]	-
U.S. NRC 10 CFR 20	Standards for Protection Against Radiation [63]	-
U.S. NRC 10 CFR 100	Reactor Site Criteria [65]	
U.S. NRC NUREG-0800	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition [66]	-
U.S .NRC NUREG-1736	Consolidated Guidance: 10 CFR Part 20 — Standards for Protection Against Radiation [67]	-
U.S. NRC Regulatory Guides (RG) 1.8	Qualification and Training of Personnel for Nuclear Power Plants [68]	Revision 4 (2021)
U.S. NRC RG 1.13	Spent Fuel Storage Design Basis	Revision 2 (2007)
U.S. NRC RG 1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste [69]	Revision 3 (2021)
U.S. NRC RG 1.45	Guidance on monitoring and responding to reactor coolant system leakage [70]	Revision 1 (2008)
U.S. NRC RG 1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants [71]	Revision 1 (2009)
U.S. NRC RG 1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I [72]	Revision 1 (1977)
U.S. NRC RG 1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants [73]	Revision 0 (2003)
U.S. NRC RG 1.196	Control Room Habitability at Light-Water Nuclear Power Reactors [74]	Revision 1 (2007)
U.S. NRC RG 1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors [75]	Revision 0 (2003)
U.S. NRC RG 8.8	Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable [76]	Revision 3 (1978)
U.S. NRC RG 8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable [77]	Revision 2 (2016)

In the UK, Radiological Protection is governed by the IRR17 [15]. A comparison of the UK and US legislative framework for Radiological Protection in nuclear power generation [43] highlighted areas where the regulations differ with the potential to impact compliance of the design of the generic SMR-300 with UK legislative requirements. The impact of the differences in the regulations on the design is assessed within the document, which allows actions to be taken within the design process to address the impacts and ensure the generic SMR-300 design meets UK requirements.

The SMR-300 IRR17 Compliance Matrix [45] presents the evidence for where the design meets UK legislation and guidance relevant to Radiological Protection. It also highlights areas where the design is not yet compliant but capable of being so, for example, where the SMR-300 Design Standard for Radiological Protection [44] includes requirements that will be incorporated into the design but have not yet been completed, and areas where the design is not compliant, where the Design Reference Point (DRP) [78] and other supporting documentation are not currently able to demonstrate that the design meets the requirements.

Table 8 highlights the key differences between UK and US Regulatory requirements [43] with regards to IRR17, it explains how those differences have been addressed for Step 2 GDA.

The most significant differences between UK and US requirements in terms of UK and global deployability of the reactor design were with respect to dose limitation and area designation.

The terms ALARA, ALARP, SFAIRP and waste generation optimisation (known as Best Management Practices (BMP) and BAT amongst other things around the world) all strive to optimise impacts. BAT / BMP is used to prevent the unnecessary creation of radioactive waste or discharges, minimise the quantity and activity of any radioactive waste that is created, and minimise the impact of discharges on people and the environment. Radiation Protection must consider all aspects in optimising exposures to OSW, MoP and the environment during normal operation, AOOs and accident conditions. To ensure exposures are truly optimised, ALARA / ALARP must be considered alongside BMP/BAT/UK ALARA in a holistic manner, and the SMR-300 Design Standard for Radiation Protection [44] provides guidance for encompassing the optimisation principle within the design process to ensure the generic SMR-300 design meets requirements.

Table 8: Key differences between US / UK Regulations

[REDACTED]

Argument 2.1.1.1 – A2: Significant gaps which would affect the design and operation of the SMR-300 have been raised as risks.

10.5.3 Evidence for Claim 2.1.1.1 – A2

Risk Management Plan [79]

This document discusses the risk management process during the SMR-300 GDA. Radiological Protection risks are identified on the Holtec Britain GDA Risk Register, which can be found on the Holtec Britain SharePoint. As the project moves beyond GDA Step 2, the risks will be managed in accordance with the risk management process.

Holtec SMR-300 GDA Capturing and Managing Commitments, Assumptions and Requirements [80]

Deliverables produced as part of GDA Step 2 activities include Commitments as a means of defining areas for further consideration. These have been sentenced and, where appropriate, taken forward into the Commitments, Assumptions and Requirements (CAR) Register following the Holtec SMR-300 GDA Capturing and Managing Commitments, Assumptions and Requirements process.

10.5.4 Narrative for Claim 2.1.1.1 – A2

Radiological Protection Step 2 deliverables have compared US and UK legislative requirements and RGP and have identified where additional substantiation may be required to meet UK expectation. These deliverables include:

- UK / US Radiological Protection Regulatory Framework Comparison [43].
- SMR-300 IRR17 Compliance Matrix [45].

Any differences between UK and US standards have been identified as risks following the Risk Management Plan [79]. Where appropriate, these risks have been escalated to Commitments through the CAR process [80]. Only one risk has been identified as requiring a Commitment, relating to HVAC design, as identified in the Design Challenge: HVAC Architecture, Design Codes and Design Basis [81]**.** Details of the Commitment can be found in sub-chapter 10.15.3.

The evidence above supports the claim that any significant gaps which would affect the design and operation of the SMR-300 have been raised as risks. Where open risks exist, the close out of these is mentioned in Appendix D.

Argument 2.1.1.1 – A3: Resolutions of regulatory gaps have been identified for normal operations.

10.5.5 Evidence for Claim 2.1.1.1 – A3

UK / US Radiological Protection Regulatory Framework Comparison [43]

The UK / US Radiological Protection Regulatory Framework Comparison highlights significant differences in regulations relating to occupational dose limits and the designation of controlled or supervised areas. Other gaps have been identified as demonstrated in Table 8. This comparison between regulatory requirements and guidance resulted in the identification of a

number of risks and issues, several of which were able to be closed out through the production of the SMR-300 Design Standard [44] which is consistent with international best practice.

SMR-300 IRR17 Compliance Matrix [45]

The IRR17 Compliance Matrix compares the design of the SMR-300 against the requirements of IRR17 [15], its associated ACoP [60], Radiological Protection related SAPs [61] and associated TAGs -043 [82], -002 [83], -004 [84], -038 [85], -041 [86], 005 [87], -021 [88], -022 [89], -023 [90], -027 [91], -035 [92], where the design is not currently compliant with these requirements it was marked up as either capable of being compliant (where application of the Design Standard for Radiation Protection would enforce compliance in the future) or not compliant (where further measures or development of the design is required to achieve compliance).

SMR-300 Design Standard for Radiation Protection [44]

Regulatory gaps highlighted in the UK / US Radiological Protection Regulatory Framework Comparison, especially those where the US requirements differ from international standards: IAEA Generic Safety Requirements (GSR) Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards (BSS) [93]; Council Directive 2013/59/EURATOM Laying Down Basic Safety Standards for Protection Against the Dangers Arising from Exposure to Ionising Radiation [94] (Euratom BSS Directive); and, 2007 Recommendations of the International Commission on Radiological Protection (ICRP), ICRP-103 [95, 95] have been addressed within the SMR-300 Design Standard for Radiological Protection [44].

SMR-300 Dose Management Strategy [46]

Regulatory gaps highlighted in the UK / US Radiological Protection Regulatory Framework Comparison, especially those where UK requirements differ from international standards, have been addressed within the SMR-300 Dose Management Strategy [46].

10.5.6 Narrative for Claim 2.1.1.1 – A3

The regulatory gaps highlighted in the UK / US Radiological Protection Regulatory Framework Comparison have been addressed within the SMR-300 Design Standard for Radiological Protection [44] and the SMR-300 Dose Management Strategy [46], with dose limits and area designations aligning with UK legislation.

The differences between the ALARA principle applied in the US and globally and the ALARP principle applied in the UK are also addressed in the SMR-300 Design Standard for Radiation Protection [44]. See also discussion within Part A Chapter 5 [29].

The initial design of the SMR-300 up to the DRP was based on the SMR-160 Design Standard for Radiation Protection [96], therefore, as a result of the delta between the two design standards, the DRP does not currently comply with UK regulatory requirements in some areas. However, through application of the SMR-300 Design Standard (which supersedes the SMR-160 Design Standard) and the Dose Management Strategy, as the reactor design develops these gaps should be capable of being closed out in future IRR17 compliance audits. A number of regulatory requirements were marked as not applicable at this design stage as they are related to decisions to be made by the future site operator. Aspects of the design that are

not currently compliant with UK regulatory standards and will not be closed out by implementation of the Design Standard have been marked up as non-compliant. This does not mean that the design is not capable of becoming compliant, but that application of the Design Standard alone will not be sufficient, and further design specifications and modifications will be necessary as the design progresses to ensure compliance prior to construction and operation.

The open risks and issues all have containment actions and resolution plans in place. Each of these risks and issues are capable of being addressed fully through further development of the design.

Argument 2.1.1.1 – A4: SAPs RP1-7 will be utilised in the development of design and procedures and in the assessment of the design to demonstrate the application of ALARP.

10.5.7 Evidence for Claim 2.1.1.1 – A4

SMR-300 IRR17 Compliance Matrix [45]

The generic SMR-300 design has been considered against the legislative requirements in IRR17 [15] and associated ACoP [60] ONR SAP targets RP1-7, NT1-3 and ECR1-2 [61], relevant ONR TAGs-043 [82], -002 [83], -004 [84], -038 [85], -041 [86], 005 [87], -021 [88], -022 [89], -023 [90], -027 [91], -035 [92], and IAEA SSG-90 guidance [62] within the SMR-300 IRR17 Compliance Matrix [45].

10.5.8 Narrative for Claim 2.1.1.1 – A4

The SMR-300 IRR17 Compliance Matrix [45] assesses the design of the generic SMR-300 against UK legislation, including SAPs RP1-7 [61]. The matrix provides evidence that shows that the design is either already compliant or capable of demonstrating compliance for each of the RP SAPs. The evidence is summarised here:

- [REDACTED]

Argument 2.1.1.1 – A5: Documentation and assessments will be developed following the ACoP to IRR17.

10.5.9 Evidence for Claim 2.1.1.1 – A5

SMR-300 IRR17 Compliance Matrix [45]

The SMR-300 IRR17 Compliance Matrix assesses the design of the generic SMR-300 against the legislative requirements in IRR17 and its associated ACoP [60].

SMR-300 Dose Management Strategy [46]

The Dose Management Strategy was produced to ensure that the SMR-300 design complies with IRR17 and associated ACoP [60]. It also addresses the requirement to produce policies and procedures in accordance with IRR17.

SMR-300 Design Standard for Radiation Protection [44]

To ensure international consistency in support of developing a fleetwide design, the standard is based on ICRP-103 [95], the IAEA BSS [93], EURATOM BSS Directive [94], and IAEA SSG-90 [62], addressing IRR17 [60] requirements, the associated ACoP and guidance [60] were consulted to ensure that UK codes and standards were considered. Where the RGP was UK centric this was noted and incorporated into the Dose Management Strategy [46].

10.5.10 Narrative for Claim 2.1.1.1 – A5

The SMR-300 IRR17 Compliance Matrix [45] provides evidence that shows that the design is either already compliant or capable of demonstrating compliance with the IRR17 regulations and ACoP [60].

There are four IRR17 regulations where insufficient information is available at Step 2 to determine compliance of the design. These are summarised in Table 9.

Table 9: IRR17 Regulations where compliance not demonstrated at GDA Step 2

[REDACTED]

The Dose Management Strategy [46] details the requirements for the development of strategies, policies and procedures as the design develops, and future operator responsibilities, it also provides methodologies for the assessment of dose to OSW and MoP as well as describing the strategy and methods to be applied in shielding assessments.

10.5.11 CAE Summary

The claim that design of the SMR-300 is compliant with Radiological Protection regulatory requirements is **partially met** at this stage of the SMR-300 design. The generic Holtec SMR-300 is shown to be broadly capable of being compliant with UK Radiological Protection legislative requirements. This is demonstrated by the following:

- A review of relevant UK legislation against US legislation to ensure all UK requirements are met or surpassed [43].
- A review of the design against IRR17, its ACoP requirements and SAPs RP1-7 [45].

Any gaps between US and UK requirements have, in the first instance, been addressed in the Dose Management Strategy [46] or the SMR-300 Design Standard for Radiation Protection [44]. Any other gaps which would affect the design and operation of the SMR-300 have been raised as risks and will be managed as part of the Risk Management Plan [79].

10.6 MANAGING DOSES

Claim 2.1.1.2: Doses associated with the generic SMR-300 are below legal limits and are demonstrated to be ALARP.

Claim 2.1.1.2 has been further decomposed into five arguments to address the claim.

- Dose targets, constraints and limits are identified for normal operations (A1).
- Methods for assessing doses to OSW and MoP during normal operations are derived (A2).
- Doses to OSW and MoP during normal operations are demonstrated to be tolerable and ALARP (A3).
- A UK legislation compliant radiation and containment zoning scheme has been developed for the UK SMR-300 (A4).
- RGP will inform the approach to ensure radiological hazards and doses are reduced to ALARP (A5).

Argument 2.1.1.2 – A1: Dose targets, constraints and limits are identified for normal operations.

10.6.1 Evidence for Claim 2.1.1.2 - A1

SMR-300 Design Standard for Radiation Protection [44]

The SMR-300 Design Standard for Radiation Protection specifies the dose constraints and limits applicable for OSW and MoP during normal operations, including AOOs, as outlined in the SMR-300 Dose Management Strategy [46]. These are presented in Table 10 together with the equivalent Basic Safety Objective (BSO) and Basic Safety Level (BSL) from the SAPs [61] where applicable.

SMR-300 Dose Management Strategy [46]

The SMR-300 Dose Management Strategy derives the dose targets, constraints and limits based on legislative requirements and OPEX from the existing international Pressurised Water Reactor (PWR) operating fleet.

Table 10: Dose Constraints and Targets

ONR Target	Category	Dose Constraint	BSO	BSL
Target 1 - Normal Operations Dose Targets – Individuals on Site	Individual Dose at Power	1 mSv/y	1 mSv/y	20 mSv/y
	Individual Outage Dose	2 mSv per task 10 mSv/y	1 mSv/y	20 mSv/y
Target 2 - Normal Operations Dose Targets – Groups on Site	Group Average Outage Dose	2 mSv/y	0.5 mSv/y	10 mSv/y
	Site Collective Dose	150 person-mSv/y	-	ALARP Based
	Other Site Workers	0.5 mSv/y		1 mSv/y
Target 3 - Normal Operations Dose Targets – Any Person off Site (UK)	Public Dose	0.02 mSv/y	0.02 mSv/y	1 mSv/y

10.6.2 Narrative for Claim 2.1.1.2 - A1

The dose limits specified in the SMR-300 Design Standard for Radiation Protection [44] are compliant with IRR17 [60] and the BSL [61], which are presented in the Dose Management Strategy.

'The dose constraints closely align with the BSO which will drive the design towards lower doses, ensuring ALARP, and the Dose Management Strategy provides the rationale behind their selection [46].

Any new facility or activity should at least meet the BSLs, where the BSL is also a legal limit then this must not be exceeded. However, even if the BSLs are met, the risks may not be ALARP and the designer is obliged reduce the risks further. Deciding when the level of risk is ALARP needs to be justified on a case-by-case basis, applying the legal test of gross disproportion. A graded approach should be used so that the higher the risk (or hazard), the greater the degree of disproportion applied, and the more robust the argument needed to justify not implementing additional safety measures.

The BSOs form benchmarks that reflect modern safety standards and expectations. The BSOs also recognise that there is a level beyond which further consideration of the safety case would not be a reasonable use of ONR resources, compared with the benefit of applying these resources to areas of higher risk. ALARP considerations, however, may be such that achieving doses above the BSO are justified, or likewise, if it is reasonably practicable to provide a higher standard of safety, then this is required by law.

Dose constraints have been set at the design stage for planning or design purposes. It is an upper level of individual dose which are not expected to be exceeded in a well-managed workplace. It is an important tool in ensuring that individual exposures are restricted SFAIRP. Dose constraints have been derived at values that are aspirational but achievable and are generally set based on OPEX from similar past work. Where dose assessment demonstrates that doses exceed the dose constraint, control measures shall be reviewed, as part of an ALARP review or optioneering study to identify alternative design options which may result in predicted doses below the dose constraint. Constraints may be set for individual tasks (recommended) or may be set across broad groups of the workforce (less valuable). In many cases, these design constraints may exceed the BSO value, especially for tasks where elevated dose rates may not be reasonably practicably mitigated. Design constraints have been based on feasibility and practicalities of implementing safety systems during the design phase. Nevertheless, even when higher than the BSO, doses at the dose constraint can still be demonstrated to be ALARP.

Argument 2.1.1.2– A2: Methods for assessing doses to workers and members of the public during normal operations are derived.

10.6.3 Evidence for Claim 2.1.1.2 – A2

SMR-300 Dose Management Strategy [46]

The SMR-300 Dose Management Strategy defines the methodology for producing a Normal Operations Dose Assessment (NODA), including dose calculation methods and formulae. It notes that at the PSR stage, there is insufficient information in terms of dose rate, occupancy, and manning levels to be able to produce a NODA at Step 2 GDA. The Dose Management

Strategy also provides a methodology and preliminary assessment of dose to MoP from direct radiation exposure.

10.6.4 Narrative for Claim 2.1.1.2 – A2

A detailed dose assessment methodology is provided in Appendix A of the Dose Management Strategy [46]. The key steps that will be undertaken to produce the NODA are:

- Define assessment objectives.
- Identify sources of radiation.
- Establish exposure scenarios.
- Select dose assessment methodology.
- Calculate dose estimates.
- Evaluate limits.
- Advise on design and / or operational improvements to optimise doses.

The dose calculation methods and formulae are provided in Appendix B of the Dose Management Strategy [46] and cover planned work dose assessment including effective dose, equivalent dose, and committed effective dose.

The methodology and preliminary assessment of dose to MoP from direct radiation are provided in Appendix D of the Dose Management Strategy [46].

Argument 2.1.1.2 – A3: Doses to workers and members of the public during normal operations are demonstrated to be tolerable and ALARP.

10.6.5 Evidence for Claim 2.1.1.2 – A3

Dose Management Strategy [46]

The SMR-300 Dose Management Strategy defines the methodology for producing a NODA and describes the process for optimising the design. It also provides a methodology and preliminary assessment of the dose to MoP from direct radiation exposure.

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard for Radiation Protection defines the process for undertaking ALARP/ALARA reviews and includes an ALARP/ALARA checklist in Appendix B.

10.6.6 Narrative for Claim 2.1.1.2 – A3

The SMR-300 Dose Management Strategy [46] defines the methodology to produce a NODA, including dose calculation methods and formulae, noting that at PSR stage there is insufficient information in terms of dose rate, occupancy and staffing levels to be able to produce a meaningful NODA at Step 2 GDA. NODA will be completed once provisional data is available to support the development of the design and will be updated as the design progresses to ensure that doses are optimised and risks reduced ALARP.

The NODA is an integral process for optimising the design, managing safety and regulatory compliance of the SMR-300 throughout its lifecycle. Assessment of doses to MoP also supports the demonstration of BAT. The output from the NODA is compared with the dose constraints established in the Radiation Protection Design Standard [44], to ensure that

assessed doses are below the site dose constraints. ALARP optimisation has been considered in the process of establishing the dose constraints for the site.

The NODA is used to support ALARP assessments. As part of the SMR-300 design development, an ALARP review will be undertaken for every process and task involving radiation exposure. The analysis of each system and process subject to ALARP assessment will be documented. Where a system is exempted from ALARP review, the rationale will be documented. The site operator will continue to apply ALARP reviews at each phase of the SMR-300 lifecycle.

The Dose Management Strategy [46] presents an initial scoping assessment of dose to MoP from direct radiation exposure to the reactor and spent fuel storage in support of PER Chapter 3 [3], based on the site parameters set out in the generic site description presented in Chapter A2 [8].

The dose to the representative person for the site from direct radiation was calculated in the Radiological Impact Assessment Topic Report [97] to be [REDACTED], conservatively assuming they live 100 m from the reactor and 80 m from the ISFSI [46], and remain at that location all year. A distance of 80 m was chosen as dose rate data was available for this distance, the precise location of the ISFSI is not included within the Generic Site Envelope [98], and a conservative assumption regarding location of reactor, ISFSI and receptor distances was desired to ensure a dose that would be bounding of any potential future site. This assessment is based on measured dose rates at this distance from a Holtec UMAX ISFSI in the US. As the site layout is developed, especially once a site has been selected, the location of the ISFSI in relation to OSW and off-site MoP will be optimised to ensure that doses to both receptor groups is less than 20 μSv/y and ALARP.

Argument 2.1.1.2 – A4: A UK legislation compliant radiation and contamination zoning scheme has been developed for the UK SMR-300.

10.6.7 Evidence for Claim 2.1.1.2 – A4

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard for Radiation Protection defines a radiation and contamination zoning scheme for SMR-300 which has been developed based around typical schemes in the US, UK and Europe, incorporating specific regulatory requirements including those set out in IRR17 [15].

Dose Management Strategy [46]

The Dose Management Strategy presents the zoning schemes established in the Design Standard for Radiation Protection [44] as would be applied in the UK to enable the assessment of the design of the SMR-300 against UK regulatory requirements.

10.6.8 Narrative for Claim 2.1.1.2 – A4

The requirements for designating areas as controlled or supervised are defined in the international safety standards: IAEA BSS [93], IAEA SSG-90 [62] and ICRP-103 [95]. The designations of supervised and controlled areas in the UK align with those in the EURATOM

BSS Directive [94] and IAEA BSS [93]. The UK incorporates these standards into its legal framework through IRR17 [15].

SSG-90 [62] recommends that at the design stage, controlled areas within a nuclear power plant are divided into zones on the basis of anticipated radiation levels and contamination levels. In accordance with SSG-90, the SMR-300 preliminary high level radiation zoning has been undertaken, based around the proposed zoning scheme, to assist with the layout design of the plant. An example of the high level preliminary radiation zoning undertaken for the SMR-300 is presented in Dose Management Strategy [46]. Details on the zoning schemes are covered in sub-chapter 10.13 and presented in Appendices A and B.

It is the responsibility of future UK operators to implement their own zoning scheme with the understanding that basic requirements around achieving controlled and supervised area requirements are met in the design.

Argument 2.1.1.2 – A5: RGP will inform the approach to ensure radiological hazards and doses are reduced to ALARP.

10.6.9 Evidence for Claim 2.1.1.2 – A5

SMR-300 IRR17 Compliance Matrix

The ALARP procedural checklist in Table 2 and the ALARP implementation checklist in Table 3 of the Compliance Matrix assess aspects of the ALARP design process and implementation in alignment with UK regulations and RGP [45].

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard for Radiation Protection incorporates the guidance provided in the IRR17 ACoP [60], SSG-90 [62], Nuclear Energy Agency Occupational Radiological Protection Principles and Criteria for Designing New Nuclear Power Plants [99] and the Nuclear Industries Guide to Changeroom design, Operation and Maintenance [100] .

Dose Management Strategy [46]

The Dose Management Strategy supports the Design Standard for Radiation Protection [44] by providing the discussion on how the Design Standard will be implemented in the UK, as documented in the ACoP and Guidance [60].

10.6.10 Narrative for Claim 2.1.1.2 – A5

The initial design of the SMR-300 up to the DRP was based on the SMR-160 Design Standard for Radiation Protection [96]. The SMR-160 Design Standard was initially only produced to meet US national regulatory requirements and considered the 2005 revision of IAEA Safety Series Guidance Radiation Protection Aspects of Design of Nuclear Power Plants [101]. To ensure international consistency in support of a fleet wide design, the SMR-300 Design Standard for Radiation Protection was produced based on guidance provided in SSG-90 [62] including dose limitation and optimisation, methodologies for calculating on and offsite radiological conditions and the measures to be taken in the design for protection of OSW and MoP. The SMR-300 Design Standard will be applied to the design and, as a result, the design will address UK specific requirements of IRR17 [15].

The terms ALARA used globally, ALARP (and SFAIRP) used in the UK, and waste generation optimisation (known as BMP and BAT amongst other things around the world), all strive to optimise impacts. BAT / BMP is used to prevent the unnecessary creation of radioactive waste or discharges, minimise the quantity and activity of any radioactive waste that is created, and minimise the impact of discharges on people and the environment. To ensure exposures are truly optimised, ALARA / ALARP shall be considered alongside BMP / BAT / UK ALARA in a holistic manner.

Application of optimisation principles is achieved in the SMR-300 design by establishing design dose constraints, or targets based on OPEX and best nuclear industry practices as identified in ICRP-103 [95] and as generally required in Radiological Protection legislation worldwide.

The SMR-300 Design Standard for Radiation Protection [44] provides an ALARA / ALARP checklist applicable for reviewing the design of the SMR-300 thus contributing to design optimisation compiled from a number of sources including Nuclear Energy Agency [99] and UK Nuclear Industry Good Practice Guide [102].

The Dose Management Strategy [46] discusses how the Design Standard has been implemented in the design of the SMR-300 to adhere to relevant UK legislation and guidance, including: IRR17 [60]; ONR SAPs [61]; ONR TAGs -002 [83], -038 [85] and -043 [82] and IAEA SSG-90 [62], while also considering OPEX from other similar operating reactors. Where this approach has not already been implemented the dose management strategy defines the process to be applied as the design progresses. NODA is an iterative process, carried out at each stage of design to inform the development of SSCs. Through this iteration process, the NODA is able to effect change on the design at the earliest possible stage to ensure risks are ALARP.

10.6.11 CAE Summary

The claim that doses associated with the generic SMR-300 are below legal limits and are demonstrated to be ALARP is partially met at this stage of the SMR-300 design. At this stage, assessment of dose to OSW has not been completed; however, preliminary assessment of doses to MoP [97] demonstrate that doses are well below the legal limit and are capable of becoming ALARP as the design develops.

The Dose Management Strategy [46] and the Design Standard for Radiation Protection [44] define the dose targets, constraints, limits and methods for assessing doses to OSW and MoP during normal operations. They also define the process to be used to produce the NODA and undertake ALARP studies. NODA production and ALARP studies have not been undertaken at PSR V1 due to limited design, dose rate and occupancy data. However, RGP will be considered when such studies are undertaken to ensure radiological hazards and doses are reduced to ALARP. In accordance with the SMR-300 Design Standard UK legislation compliant radiation and contamination zoning schemes have been developed which are presented in Appendix A and Appendix B, respectively and shall be applied to the design.

The SMR-300 Design Standard incorporates international best practice for Radiological Protection in nuclear facilities and provides a comprehensive ALARA / ALARP checklist to ensure that doses and risks to OSW and MoP will be ALARP. Application of this Design

Standard as part of ongoing development of the design of the SMR-300 will ensure that doses and risks will be optimised.

10.7 HIERARCHY OF CONTROLS MAKES RISKS ALARP

Claim 2.1.1.3: The safety hierarchy of controls will be applied to ensure radiological hazards will be reduced to ALARP.

Claim 2.1.1.3 has been further decomposed into two arguments to address the claim.

- Sources of radiological risk or dose are eliminated within the design where possible (A1).
- Where sources cannot be eliminated, radiological risk will be reduced through minimising the likelihood and severity of events that will give risk to radiological risk (A2).

Argument 2.1.1.3 – A1: Sources of radiological risk or dose are eliminated within the design where possible.

10.7.1 Evidence for Claim 2.1.1.3 - A1

SMR-300 IRR17 Compliance Matrix [45]

The IRR17 Compliance Matrix evidences the steps taken to apply the HoC and eliminate the likelihood and severity of radiological risk. Utilising the HoC ensures that the most effective measures are prioritised to protect workers from hazards. Examples within the design that demonstrate these principles are presented in Appendix B, Table 4.

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard states that exposures will be prevented or minimised through optimisation, application ERICPD and the application of HoC measures.

SMR-300 Dose Management Strategy [46]

The Dose Management Strategy describes the structured, hierarchical approach to the management of risks whereby risks are reduced using the ERICPD approach. This is aligned with the IRR17, HoC for managing radiation risks (Regulation 9) [15]. It also provides an overview of selected approaches for dose elimination.

10.7.2 Narrative for Claim 2.1.1.3 - A1

The IRR17 Compliance Matrix identifies examples of controls in the current design that eliminate the radiological risk:

- Motor-operated valves and automatic valves within passive containment heat removal system [103], and containment ventilation system [104].
- Ability to mitigate design basis accidents with no operator action [103].
- Ventilation systems will be designed to ensure that contaminated areas are at lower pressure than the general areas. Air flow will be directed from the general area to the more contaminated areas [105].
- The CS shall prevent or limit the release of radioactive material in all operational states and design basis accident conditions throughout the life of the plant [106].

- The RAB is designed to provide remotely operated process instrumentation and control for radiation protection for plant personnel [107].
- The RAB is designed to isolate substantial sources of radiation [107].
- Whenever possible, pipes with radioactive fluids shall be routed through pipe chases or shielded areas, and away from pipes for "clean" services. The large pressuriser eliminates the need for power operated relief valves which could be a source of leakage [107].

The SMR-300 Radiation Protection Design Standard [44] provides guidance on design features, guided by the HoC, to protect OSW and MoP for all reactor lifecycle phases.

The predominant way to ensure radiation doses to OSW and MoP meet limits and targets, and doses are optimised is to reduce the hazard SFAIRP using the ERICPD methodology, and the adoption of good engineering practice in the design features of the plant. Protection shall be optimised through the implementation of the HoC measures, such that passive and engineered controls, including shielding and physical access controls, design features and safety features are prioritised, supported by active and administrative controls and PPE.

The Design Standard [44] provides examples of elimination of risks including:

- Systems and components shall be designed for high reliability and maintainability to reduce or eliminate the need for repair or preventive maintenance on radioactive components.
- Designers shall adopt a plant layout and introduce engineered features that together ensure the use of personal protection equipment is minimised to the extent practicable. Where it is not reasonably practical to eliminate or control a respiratory radiological hazard in the plant, the design documentation shall indicate that suitable respirators are needed to protect OSW from the hazard.

The Dose Management Strategy [46] provides the strategy to eliminate the radiation hazard in the SMR-300. The following are examples of measures that have been incorporated into the design to ensure the removal of either the source of radioactivity or the need for human intervention:

- Radioactive waste and fission product removal.
- Passive safety features, such as motor-operated valves, rely on passive safety systems driven by gravity and natural circulation, reducing reliance on active components and human intervention.
- Selecting materials with minimal activation potential.

Argument 2.1.1.3 – A2: Where sources cannot be eliminated, radiological risk will be reduced through minimising the likelihood and severity of exposure to radiological risk.

10.7.3 Evidence for Claim 2.1.1.3 – A2

SMR-160 Design Standard for Decommissioning [108]

The Design Standard for Decommissioning describes the approach for minimising the radiation source specifically addressing fuel design, selection of materials, reduction and spread of contamination.

SMR-300 IRR17 Compliance Matrix [45]

The IRR17 Compliance Matrix evidences the steps that have been taken apply the HoC and either eliminate or minimise the likelihood and severity of radiological risk. Utilising the HoC ensures that the most effective measures are prioritised to protect workers from hazards. Examples within the design which demonstrate these principles are presented in Appendix B, Table 4.

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard addresses the entire lifecycle and states that exposures will be prevented or minimised through optimisation and application of ERICPD and the application of HoC measures.

Piping and Equipment Layout Guidelines for Ensuring Radiation Exposures ALARA [109]

This Piping and Equipment Layout Guidelines defines the approach to be adopted for shielding of equipment compartments and cubicles to meet zoning requirements. It also outlines the requirements for layout and design of piping, valves and components.

SMR-300 Dose Management Strategy [46]

Where elimination is not possible, implementation of the HoC measures emphasise the importance of implementing engineering controls such as shielding, interlocks and containment, over administrative controls such as work planning and access restrictions. A final layer of defence is the use of PPE. This is consistent with the requirements in Regulation 9 of IRR17 [15].

Nuclear Island Minimization Strategy for Activity Generation and Accumulation [110]

This report presents the SMR-300 strategy to minimise activation and corrosion product generation and accumulation. This report highlights the strategies used to minimise the elements that can be activated that contribute to higher out of core dose rates. It also highlights the strategies in the SMR-300 design that minimise the accumulation of these activated products in the plant.

Part B Chapter 23 [27]

Evidence for Claim 2.2.14.4 describes how the chemistry of NI systems reduces the normal operation source term of the RCS.

Part B Chapter 2 [9]

Claim 2.2.3 Adequate provision for the control of radiation exposure and control of release of radioactive material is incorporated into the design of the reactor systems, supporting facilities, engineered safety features, and fuel and core design.

10.7.4 Narrative for Claim 2.1.1.3 – A2

The SMR-160 Decommissioning Design Standard [108] specifies that proper materials and processes should be chosen to minimise the generation of radioactive waste, plant layouts should be designed to limit the spread of contamination, plant layouts should facilitate access

for decommissioning, and interim and final waste storage facilities should be given consideration.

The design features that were considered in the design of the SMR-160 and have been or shall be considered in the SMR-300 design are:

- Reduction of the radiation source e.g. fuel design, material selection and reduction in surface contamination.
- Plant layout designed to limit the spread of contamination, e.g. positioning of floor drains, segregation of non or lesser contaminated areas from higher contaminated areas, airflows from higher to lower areas of radioactivity.
- Facilitate the dismantling and decontamination of radioactive equipment
    - design and placement of pipes and ducts should allow easy access for cleaning and removal.
    - sumps, tanks and drains should prevent spread of contamination and facilitate clean up.
    - crud traps should be considered.
    - ventilation systems should control and minimise contamination spread.
    - Systems shall be able to be isolated from contaminants.
- Simplification of waste management e.g. minimising the volume of radioactive material requiring disposal and utilising direct access routes to avoid clean areas.
- Design the plant for storage with surveillance or deferred dismantling e.g. maintaining and monitoring the plant for a period that allows decay.

The IRR17 Compliance Matrix, when comparing the design against Regulation 9 of IRR17 [15] and associated ACOP [60], identifies examples of controls in the current design that minimise the radiological risk:

- [REDACTED]
- The RAB will incorporate design features to minimise accumulation of radioactive materials in resin and sludge treatment systems etc. [59].
- Access inside the containment structure is prevented during normal operation. [66]
- If a hazardous condition exists behind the access, the physical barrier over the access should be equipped with an interlock that will de-energise the hazardous equipment when the barrier is opened or removed [111].
- A radiologically controlled area (RCA) associated with higher dose rates should be kept as small as reasonably achievable consistent with accessibility for accomplishing the services that must be performed in those areas, including equipment laydown requirements. RCA where plant personnel spend substantial time should be designed to the lowest practical dose rates [105].
- The SMR-300 incorporates a passively safe system that relies on natural phenomena meaning there are fewer active components to maintain or repair, reducing time spent in radiological area and number of staff required to operate [46].
- The SMR-300 components like the reactor core, steam generators, pressuriser and primary piping are housed in a single vessel, reducing the number of leak points minimising worker access to high radiation areas by concentrating radioactive materials in a smaller, shielded volume and simplifying maintenance [46].

- SMR-300 uses materials with reduced activation potentials such as a reduction in materials containing cobalt, silver and antimony in components near the reactor core [46].
- Integration of robotics and remote-controlled systems for inspection, maintenance, and fuel handling reduces the need for manual intervention in high-radiation areas [46].

The SMR-300 Design Standard [44] provides guidance on design features, applying the HoC in accordance with IRR17, Regulation 9 of ACoP [60], to protect OSW and MoP for all reactor lifecycle phases. It is consistent with the guidance provided in SSG-90 [62]. This builds upon the SMR-160 Design Standard [96] which aligned with the 2005 revision of IAEA Safety Series Guidance Radiation Protection Aspects of Design of Nuclear Power Plants [101]. The Design Standard presents a general design approach for limiting exposures to OSW and MoP through:

- Minimising the production and build-up of radionuclides.
- Employing advanced design principles for local plant features including the plant layout, shielding design, and the design of systems and components for reliability and maintainability.
- Controlling access to areas of high activity or radiation, or where the potential for possible contamination exists.
- Provision for auxiliary features (i.e., for storage and donning of personal protective clothing) and decontamination facilities where required.
- Personnel monitoring and dosimetry facilities to demonstrate compliance with NRC regulations.
- Controlling access of unauthorised persons within the exclusion area boundary.
- Monitoring and controlling the release of all effluents, liquid and gaseous, that may conceivably carry radioactivity; and
- Storing active solid wastes in a manner that prevents radioactive release.

This approach, which has been used in the development of the design of SMR-300, aligns with the HoC principle.

The Dose Management Strategy [46] provides the strategy to minimise the radiation hazard in the SMR-300. The following examples have been incorporated into the design to minimise radiation exposure risks to OSW and MoP, and the environment:

- Corrosion prevention and radiation source term reduction.
- Shorter duration in-containment wet storage of spent fuel.
- Dry cask storage of spent fuel.
- Reducing waste accumulation and optimising processing.
- Selecting materials with lower activation potential.
- Fewer radioactive sources as a result of simplified design and fewer plant items.

Minimisation through material selection, waste accumulation and processing, and corrosion prevention and radiation source reduction, will be achieved through guidance provided in the Nuclear Island Minimization Strategy for Activity Generation and Accumulation [110]. The design of the SMR-300 will minimise the use of cobalt, silver and antimony which are the greatest contributors to out of core dose rates. Accumulation of activation products will be in targeted locations, out of high occupancy areas and be shielded to prevent inadvertent

exposure. Allowing activated products to decay for a period of time prior to transfer out of vessels will reduce short lived, high activity isotopes.

The design of the fuel will minimise the likelihood and severity of an accident. The core is being specifically designed with the IAEA barriers:

- Fuel matrix.
- Fuel rod cladding.
- Reactor coolant circuit.
- Reactor containment structure.

There is insufficient information at this stage to determine whether the layout and any dose reduction provisions would result in doses that are ALARP; however, Sub-chapter 9.6 of Part B Chapter 9 [23] outlines EIMT guidance for requirements, designing for and scheduling EIMT and comparison of UK vs USA expectations. Therefore, measures are being taken to ensure that the overall risks of the design are likely to be ALARP. Additionally, Part A Chapter 2 [8] demonstrates the top level requirements and design philosophy are appropriate for demonstrating a design is likely to be ALARP, and Part A Chapter 4 [22] presents the design controls for EIMT and layout in sub-chapter 4.8.2.

10.7.5 CAE Summary

The claim that safety HoC will be applied to ensure radiological hazards will be reduced to ALARP is principally met at this stage of the SMR-300 design. The Dose Management Strategy [46] describes the structured, hierarchical approach to the management of risks whereby risks are reduced using the ERICPD approach. This is aligned with the IRR17, HoC for managing radiation risks. The SMR-160 design standard in which the DRP has been designed against incorporated a HoC consistent with international standards. Application of the SMR-300 Design Standard, which more closely aligns with Regulation 9 of IRR17, will only strengthen the arguments and evidence for this claim.

The primary aim is to eliminate the radiation hazard by choice of materials, chemistry control (see Part B Chapter 23 [27]) and passive safety systems that require no operator action.

Where elimination is not possible, the HoC emphasises the importance of implementing engineering controls such as shielding, interlocks and containment, followed by administrative controls such as work planning and access restrictions and finally PPE.

10.8 SOURCES ARE APPROPRIATELY CHARACTERISED

Claim 2.2.15.1: Sources of residual radiological risk are appropriately characterised to inform the design of SSCs.

Claim 2.2.15.1 has been further decomposed into two arguments to address the claim:

- Derived source terms are used to define the requirements for SSCs (A1).
- Appropriate methodologies are used to calculate the best estimate and design basis SMR-300 Source terms (A2)

Argument 2.2.15.1 – A1: Derived source terms are used to define the requirements for SSCs.

10.8.1 Evidence for Claim 2.2.15.1 – A1

SMR-300 Source Terms [36]

Derives source terms for the SMR-300 fuel using the SCALE 6.2.1 computer code.

SMR-300 Fission Neutron and Gamma Sources [112]

Calculates the generation of prompt fission neutron and prompt fission gamma sources for the SMR-300 core during normal reactor operations.

Contained Radiation Sources for Normal Operations **[32]**

Derives source terms for primary and secondary coolant using data from the SMR-300 Source Term report [36] above.

SMR-300 Spent Fuel Pool Cooling System Source Terms [37]

Derives source terms for the SFP coolant, SFP filter and mixed-bed demineraliser.

Part B Chapter 23 [27]

Evidence for Claim 2.2.14.4 describes how the chemistry of NI systems reduces the normal operation source term of the RCS.

10.8.2 Narrative for Claim 2.2.15.1 – A1

Accurate determination of source terms is a critical part of system design to understand the nature of radiation hazard in the core that extends to other parts of the system. The source terms are used to ensure SSCs are designed and developed to provide a level of mitigation throughout the various parts of the plant to ensure doses to OSW and MoP are within limits and ALARP, and resulting impacts from Radioactive Waste are ALARA, through application of BAT and the Waste Hierarchy.

Source terms for SMR-300 will continue to be developed and modified as the design develops and will contribute to the design of the SSCs. The source terms that have already been developed for SMR-300 and those still under development are listed in Table 2. The source terms available reflect the current maturity of the design; however, the available source terms are considered sufficient to complete a fundamental Step 2 GDA assessment. Source terms are sufficiently developed to carry out preliminary dose rate analysis to perform radiation zoning of plant items, which will feed into shielding design, to identify risks associated with

entry to containment and to support the assessment of impacts on MoP and the environment. The developed source terms are sufficient to allow preliminary analysis of the shielding SSC requirements. The airborne source terms are necessary to complete preliminary contamination zoning of the CS and RAB, as these have not yet been finalised, it is not possible to assess this aspect of the design, which will feed into the RCV and RAB layout.

These source terms have been utilised to derive discharge source terms in PER Chapter 2 [34] which have informed the Radiological Impact Assessment [3], BAT [14] Radioactive Waste Management Arrangements [33] and Monitoring and Sampling [35] Chapters of the PER. They have also fed into preliminary shielding analysis in the Shielding Basis of Design [47] and allowed provisional radiological zoning of areas in the Dose Management Strategy [46] which will inform the development of shielding SSCs.

Argument 2.2.15.1 – A2: Appropriate methodologies are used to calculate the best estimate and design basis SMR-300 Source terms

10.8.3 Evidence for Claim 2.2.15.1 – A2

Evaluation of SMR-300 Calculated Source Terms Against Publicly Available Information [31]

The source term evaluation document [31] describes the methodologies employed to calculate source terms and evaluates the calculated source terms against publicly available data.

Part B Chapter 23 [27]

The Chemistry chapter provides arguments and evidence against Claim 2.2.14.4: The chemistry of NI systems reduces the normal operation source term of the RCS.

See also sub-chapter 10.2.2

The methods utilised to derive the source terms are summarised in sub-chapter 10.2.2 and reference out to the individual source term documents including:

- SMR Source Terms [36].
- Contained Radiation Sources for Normal Operation [32].
- SMR-300 Spent Fuel Pool Cooling System Source Terms (Preliminary) [37].
- Estimate of the SMR-300 Gaseous and Liquid Effluent Releases using the Gaseous and Liquid Effluents - Pressurised Water Reactors (GALE-PWR) 3.2’ [38].
- Estimate of the SMR-300 Gaseous and Liquid Effluent Releases using the Gaseous and Liquid Effluents - Pressurised Water Reactors (GALE-PWR) 3.2’ [38].
- Calculation of SMR-300 Solid Radiological Waste Inventories [39].
- Calculation of RAB and containment Airborne Concentrations [113].

10.8.4 Narrative for Claim 2.2.15.1 – A2

The methods utilised to derive the range of source terms produced for the SMR-300 reactor design follow US standards. Evaluation of the calculated source terms against OPEX has demonstrated that although the activity concentrations of individual radionuclides appear high when compared to observed data, they build in appropriate conservatisms for this stage of design development.

10.8.5 CAE Summary

The claim that sources of residual radiological risk are appropriately characterised to inform the design of SSCs. is principally met at this stage of the SMR-300 design. Source terms have been derived, using US standards and methods, for gamma and neutron emissions, decay heat and radiological inventory for use in shielding, normal operations dose, radiological consequence, radioactive waste management and effluent discharge impact assessments. Evaluation of these source terms against observed data from other PWRs has shown that the calculated source terms are appropriate for this stage of design. These source terms will be used to inform the design of SSCs. Source terms will continue to be progressed alongside development of the design.

10.9 SHIELDING IS DESIGNED TO MINIMISE EXPOSURES

Claim 2.2.15.2: The generic Holtec SMR-300 shielding is designed and substantiated to minimise exposures for all plant areas and operational stages, including waste package transport.

Claim 2.2.15.2 has been further decomposed into two arguments to address the claim:

- Shielding will be designed with reference to RGP to achieve or improve upon required area dose rates ensuring doses are ALARP (A1).
- Additional localised temporary shielding will be utilised for EIMT tasks as specified in task specifications (A2).

Argument 2.2.15.2 – A1: Shielding will be designed with reference to RGP to achieve or improve upon required area dose rates ensuring doses are ALARP.

10.9.1 Evidence for Claim 2.2.15.2 – A1

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard for Radiation Protection specifies the shielding design target dose rates for normal operations and discusses the shielding design basis, shielding material selection, temporary shielding and shielding multiple sources.

SMR-300 Dose Management Strategy [46]

The Dose Management Strategy presents a shielding design strategy for the generic SMR-300 addressing evaluation of shielding requirements, shielding design objectives, the shielding philosophy, shielding methodology including calculations, provisional design development and shielding strategy.

Piping and Equipment Layout Guidelines for Ensuring Radiation Exposures ALARA [109]

This report provides specific criteria and design guidelines for equipment and piping layout of the systems carrying highly radioactive fluid or equipment containing radioactive sources.

Holtec SMR-300 Shielding Basis of Design [47]

The Shielding Basis of Design presents a preliminary shielding assessment using MCNP for the SMR-300 twin unit reactor design. It provides attenuation curves for various shielding materials and shielding assessments of the reactor and SFP.

10.9.2 Narrative for Claim 2.2.15.2 – A1

There is a design intent for personnel to enter reactor containment at power to investigate faults and carry out repairs. [REDACTED]. Therefore, access at full power cannot be deemed to meet Regulation 9(1) or Regulation 12(1) under the DRP. The containment and reactive actions identified for this issue consists of identification of scenarios which would require personnel access while the reactor is on load, followed by ALARP optioneering to identify alternative methods to prevent or mitigate the risk which will ensure that doses to employees are ALARP.

Shielding design is not fully matured, provisional shielding arrangements in the DRP are based on US OPEX and area designations (which are not compliant with UK or international requirements). The SMR-300 Design Standard for Radiation Protection [44], and associated Equipment and Piping Guidelines [109] more closely align with UK and international regulatory expectations, therefore any updates to the shielding design should more closely align with UK standards.

Further work will be required post Step 2 to develop the shielding design and the NODA to demonstrate that the ALARP principles have been applied to the reactor design and operation.

The Design Standard [44] has been informed by RGP from 2007 Recommendations of the ICRP, ICRP-103 [95], IAEA BSS [93], EURATOM BSS Directive [94] and IAEA SSG-90 [62].

[REDACTED]

These targets have been selected based on OPEX and RGP.

The Dose Management Strategy [46] applies the guidance in the Design Standard [44] to meet UK specific requirements using RGP from SSG-90 [62], IRR17 [60] and the ONR Radiation Shielding TAG [83].

The Equipment and Piping Layout Guidelines [109] is based on nuclear industry design practice, regulatory guidance and Utility Required Documents (URD) design requirements as applicable to Radiation Protection and ALARA. This document provides a detailed list of major equipment and structures requiring radiation shielding, this is not an exhaustive list but will act as the starting point for the derivation of the shielding design. Many of the requirements detailed in this document are derived from the Electric Power Research Institute (EPRI) and URD requirements.

The guidelines also provide generic criteria for shielding requirements including:

- Piping expected to contain significant radiation sources should be adequately shielded and properly routed to minimise exposure to personnel.
- Major equipment in a radioactive system should be located in separate shielded compartments with short piping runs between equipment.
- Shield thickness varies depending on the source terms for each of the equipment, access/radiation zone requirements outside the cubicles/compartments.

The Shielding Basis of Design [47] provides the methods and key assumptions which will be used across all shielding assessments, defining the radioactive source data, geometry data, materials data, flux-to-dose conversion factors and assessment criteria.

The Shielding Basis of Design [47] presents preliminary dose rate calculations for the reactor inside and outside containment based on a physical model of a twin unit reactor. It also calculates dose rates external to the SFP. The model has been developed based on SMR-300 source terms for neutron and gamma emissions [36] [112] and the physical model is based on preliminary layout drawings. The report determines the maximum dose rate outside containment to be [REDACTED] which is below the limit of 0.5 µSv/h for R0 undesignated areas as defined in Appendix A. The dose rates from the SFP contents are well below background and will not contribute significantly to total dose rates inside or outside of the

containment. This work gives confidence that the proposed shielding design can achieve the dose targets.

The results from the MCNP model have been cross-checked using SCALE, which shows close agreement between the results.

A number of key assumptions and simplifications were made in the analysis which are discussed below.

[REDACTED]

Argument 2.2.15.2 – A2: Additional localised temporary shielding will be utilised for EIMT tasks specified in task specifications.

10.9.3 Evidence for Claim 2.2.15.2 – A2

SMR-300 Design Standard for Radiation Protection [44]

Where it is impracticable to provide permanent shielding, distance between components and the substantial radiation sources or temporary shields shall be used to reduce the exposure of personnel, in anticipation of maintenance and inspection that may be required during normal operation or decommissioning. Due to the compact nature of the SMR-300 design, the potential for applying distance as a dose reduction measure is reduced. The need for the use of temporary shielding, in addition to or in place of permanent shielding where permanent shielding is not practicable, shall be stated in design documentation, as an input to plant operations documents.

SMR-300 Dose Management Strategy [46]

The design shall incorporate a combination of fixed and temporary shielding solutions to manage both predictable and dynamic radiation hazards. Fixed shielding provides robust, permanent barriers for high radiation areas, while temporary shielding is used to address specific tasks such as maintenance, refuelling, or inspections ensuring operational flexibility.

Piping and Equipment Layout Guidelines for ensuring Radiation Exposures ALARA [109]

The piping guidelines provide the requirements in the EPRI URD to achieve ALARA which states basic requirements for shielding, including temporary shielding. This lists the requirement for permanent brackets for temporary shielding.

10.9.4 Narrative for Claim 2.2.15.2 – A2

The choice of temporary shielding will take account of the same factors as permanent shielding but also will consider ease of installation and removal, dose impact of installation and removal, weight and structural integrity, cost effectiveness, reusability and its environmental impact and the ease of decontamination. Consideration will be given to provision of equipment in areas where temporary shielding may be required to facilitate the installation of the shielding. Permanent frameworks are recommended to reduce installation times and storage locations close by to minimise handling of shielding.

The reactor layout shall be designed to include sufficient space for deploying temporary shielding where and when required, including space for local storage of the shielding when

not in use. This ensures operational tasks can be performed safely without compromising dose constraints or introducing significant dose accrual during the transportation and installation of shielding materials.

An ALARP review of EIMT tasks and associated temporary shielding will be undertaken to ensure that normal operations dose uptake is optimised.

10.9.5 CAE Summary

The claim that shielding is designed and substantiated to minimise exposures for all plant areas and operational stages is nominally met at this stage of the SMR-300 design. The Shielding Basis of Design [47] presents preliminary dose rate calculations based on defined source terms, preliminary shielding thicknesses and associated material properties. The report determines the maximum dose rate outside of the CES to be [REDACTED] which is below the limit of 0.5 μSv/h for R0 undesignated areas as defined in Appendix A, not taking into consideration any sources within the RAB, or the primary circuit within the CS. The dose rates from the SFP contents are well below background and will not contribute significantly to total dose rates inside or outside of the containment. This preliminary analysis gives confidence that the proposed shielding design for the CS/CES can achieve the dose targets outside of containment, including at public receptor distances. Optimisation of the shielding design will be undertaken to demonstrate that normal operation dose uptake is ALARP.

Implementation of the shielding philosophy and strategy provided in the Dose Management Strategy [46], together with the requirements in the Equipment and Piping Guidelines [109] will provide a strong foundation for the development of the shielding design to ensure that normal operation doses to both OSW and MoP will be ALARP.

10.10 MATERIALS SELECTION MINIMISES NEUTRON ACTIVATION

Claim 2.2.15.3: Material selection minimises the generation of neutron activation products in SSCs.

Claim 2.2.15.3 has been further decomposed into three arguments to address the claim:

- Source terms are appropriately characterised to inform the design of SSCs (A1).
- Low / no cobalt steels will be selected for SSCs within the biological shielding (A2).
- Selection of highly corrosion resistant materials are combined with corrosion reducing chemistry regimes to minimise the release of activatable material to primary coolant (A3).

Argument 2.2.15.3 – A1: Source terms are appropriately characterised to inform the design of SSCs.

10.10.1 Evidence and Narrative for Claim 2.2.15.3 – A1

The evidence and narrative for Claim 2.2.15.1 – A1 in sub-chapter 10.8 is identical to the argument here for Claim 2.2.15.1 – A1: Derived source terms are used to define the requirements for SSCs, and will therefore not be repeated here. The derivation of primary and secondary neutron flux within the reactor and CS will inform the requirements for the use of low activation potential materials in SSCs. This is also related to Claim 2.1.1.3 – A1: Sources of radiological risk or dose are eliminated within the design where possible, in sub-chapter 10.7. Therefore, this claim is associated with Regulation 9 of IRR 17 [15].

Argument 2.2.15.3 – A2: Low / no cobalt materials will be selected for SSCs within the biological shielding

10.10.2 Evidence for Claim 2.2.15.3 – A2

Nuclear Island Minimalization Strategy for Activity Generation and Accumulation [110]**.**

This report outlines the strategy for minimising the generation and accumulation of activation and corrosion products in the SMR-300 plant design.

10.10.3 Narrative for Claim 2.2.15.3 – A2

The Nuclear Island Minimalization Strategy [110] presents the SMR-300 strategy for material selection to minimise the generation of activation products. The SMR-300 design utilises the EPRI recommendation for maintaining cobalt content in steels to less than 0.05wt% (including in valve material). Other materials containing cobalt are also eliminated or minimised e.g. there is no use of Stellite in lower core internals. Stellite has better wear resistance than other valve seat materials, therefore requirements of valves will be evaluated prior to the selection of alternative valve seats. Materials containing cobalt, silver and antimony in components near the reactor core which give rise to out of core dose rates are minimised.

Argument 2.2.15.3 – A3: Selection of highly corrosion resistant materials are combined with corrosion reducing chemistry regimes to minimise the release of activatable material to primary coolant

10.10.4 Evidence and Narrative for Claim 2.2.15.3 – A3

Part B Chapter 23 [27] demonstrates, in 2.2.14.4-A1, that the amount of material, that is capable of becoming activated, released to the RCS is minimised.

10.10.5 CAE Summary

The claim that materials selection minimises the generation of neutron activation products in SSCs is demonstrated to be met at this stage of the SMR-300 design. The SMR-300 has been designed from the outset to ensure that materials are not susceptible to neutron activation. The Nuclear Island Minimalization Strategy [110] presents the SMR-300 strategy for material selection. The shielding philosophy for the SMR-300 is to select shielding materials designed to reduce neutron activation of surrounding structural and functional components. Consideration will be given to other elements that could become activated and pose a significant dose or waste handling / disposal-related hazard. Part B Chapter 23 [27] demonstrates that the amount of activatable material released to the RCS is minimised.

10.11 VENTILATION AND CONTAINMENT SYSTEMS DESIGN

Claim 2.2.15.4: Ventilation and containment systems are designed to meet Radiological Protection requirements and minimise exposures.

Claim 2.2.15.4 has been further decomposed into four arguments to address the claim:

- The design utilises safe-change filters to minimise the potential for the spread of contamination (A1).
- Arrangement of filter units are optimised to ensure external doses are minimised (A2).
- Ventilation systems will maintain flow rates at openings to prevent the loss of containment (A3).
- The design of ventilation systems and filter arrangements is aligned with UK relevant good practice (A4).

Argument 2.2.15.4 – A1: The design utilises safe-change filters to minimise the potential for the spread of contamination.

10.11.1 Evidence for Claim 2.2.15.4 – A1

Design Challenge: HVAC Architecture, Design Codes and Design Basis [81]

Risks associated with the design of HVAC systems were raised, and this is covered in Argument 2.2.1.1-A2 in Part B Chapter 19 [30]. The risk and process are discussed in greater detail in the Mechanical Engineering ALARP Summary Report [114].

Design Basis Report for Mechanical SSCs [115]

The Design Basis Report defines the requirements for cartridge type HEPA filters that will reduce the potential for spread of contamination during filter changes.

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard for Radiation Protection outlines the design requirements for the RCV to reduce the potential for spread of contamination during filter changes.

10.11.2 Narrative for Claim 2.2.15.4 – A1

Filter housing and internals will be designed so that operator radiation exposure during filter cartridge replacement is minimised. Cartridge filters that have high radiation levels will be handled by remote techniques. Handling procedures and equipment for filter maintenance and cartridge replacement will be standardised throughout the station to ensure that operators use the same procedures and are less likely to make errors during removal and handling.

HEPA filters, pre-filters and any charcoal or other air filter equipment will be designed for safe, contamination controlled changeout and maintenance.

The layout of rooms containing ventilation equipment will take into consideration the potential buildup of contamination within filters and equipment and be appropriately sized and shielded to minimise exposures to personnel conducting maintenance or replacement.

The ALARA Checklist presented in Appendix B of the Design Standard for Radiation Protection [44] specifies the ALARA aspects that should be considered during ventilation system design.

The current vent design does not align with UK standards for ease of change and ALARP including size reduction and packaging. Risks associated with the design of HVAC systems were raised, and this is covered in Argument 2.2.1.1-A2 in Part B Chapter 19 [30].

[REDACTED]

Gap analysis is covered in Mechanical Commitment C_Mech_028, formally captured in the Commitments, Assumptions and Requirements process [80]. This commits Holtec to conducting a detailed gap analysis to build upon the high-level gap analysis conducted during GDA to identify the differences between UK and US approaches to HVAC design.

A HVAC Working Group has been created involving topic leads from the above areas with the aim of developing a strategy for design solutions that address the requirements of all stakeholders, including both the US and UK design teams.

The output of the HVAC Working Group will be an HVAC Gap Analysis against UK RGP which will outline the further work that many be required to achieve UK RGP and demonstrate ALARP. Due to the nature and complexity of the work, the HVAC Gap Analysis will not be issued until the completion of GDA Step 2.

This risk, challenge and commitment apply to all of the ventilation claims, therefore the discussion will not be repeated for each claim.

Argument 2.2.15.4 – A2: Arrangements of filter units are optimised to ensure external doses are minimised.

10.11.3 Evidence for Claim 2.2.15.4 – A2

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard for Radiation Protection defines the design aspects that are required to ensure that external (and internal) doses are minimised by the arrangement of filter units.

See claim 2.2.15.4 - A1 for applicable risk relating to HVAC design.

10.11.4 Narrative for Claim 2.2.15.4 – A2

Section 4.2.5.3 of the Design Standard for Radiation Protection [44] defines the design aspects that should be considered to ensure that external doses are minimised by the arrangement of filter units. These aspects include:

- The layout of rooms containing ventilation equipment should take into consideration the potential buildup of contamination within filters and equipment and be appropriately sized and shielded to minimise exposures to personnel conducting examinations or maintenance.
- Filtration to be provided close to the source of contamination, where possible, in high contamination areas to minimise the build-up of contamination in ducting.

- Ventilation (including ducting) systems to minimise bends and contamination traps to reduce the dose burden during maintenance and decommissioning.

The ALARA Checklist presented in Appendix B of the Design Standard for Radiation Protection [44] specifies the ALARA aspects that should be considered during ventilation system design.

Argument 2.2.15.4 – A3: Ventilation systems will maintain flow rates at openings to prevent the loss of containment.

10.11.5 Evidence for Claim 2.2.15.4 – A3

SMR-300 Design Standard for Radiation Protection [44]

Sub-section 4.2.5, Control of Airborne Contamination and Gaseous Radiation Sources of the Design Standard for Radiation Protection states that the design shall provide protection against airborne radioactive material.

See claim 2.2.15.4 A1 for applicable risk relating to HVAC design.

10.11.6 Narrative for Claim 2.2.15.4 – A3

The airflow in the ventilation system will be such that the pressure in a region of lower airborne contamination levels is higher than the pressure in a region of potentially higher contaminations levels to ensure that air flows from low to high contamination potential and therefore minimise the transfer of contamination uncontaminated and lower contamination zones. This is surmised in the Radiologically Controlled Area HVAC System AHU and Duct Sizing [116]. Preliminary ventilation flows are provided in the Holtec Radiologically Controlled Area HVAC System Duct Routing Sketch Notes [117] which indicates flows from corridors and general areas to individual rooms. The pressure of rooms located in controlled areas will be maintained below atmospheric to prevent radioactive releases into the atmosphere during operation, taking into consideration all penetrations.

The Design Standard for Radiation Protection [44] defines the design aspects to ensure ventilation systems will maintain flow rates at openings to prevent the loss of containment. These include the following:

- Use of physical containment as the most effective means of preventing the spread of active material, minimising the number and size of openings where possible.
- Provide air flow patterns to maintain air pressure gradients and airflows from areas of low potential airborne contamination to areas of higher potential contamination to limit the spread of airborne contamination within the plant during normal operations, unplanned events, and accident conditions. Use of airlocks between areas of low and high airborne potential contamination.

The ALARA Checklist in Appendix B of the Design Standard for Radiation Protection [44] states that a system of differential pressure should be used to direct the flow of any airborne radioactive material that escapes containment and identifies design aspects that should be considered to maintain containment air flow.

The design challenge, and associated commitment related to HVAC will be further progressed beyond Step 2 GDA. See also narrative for Claim 2.2.15.4 – A4.

Argument 2.2.15.4 – A4: The design of ventilation systems and filter arrangements is aligned with UK relevant good practice.

10.11.7 Evidence for Claim 2.2.15.4 – A4

Design Challenge: HVAC Architecture, Design Codes and Design Basis [81]

The Design Challenge for HVAC Architecture, Design Code and Design Basis provides a strategy to apply UK guidance / RGP into the development of the HVAC design. This will ensure the HVAC design supports the reduction of risks to ALARP for a UK deployed SMR-300.

SMR-300 Design Standard for Radiation Protection [44]

The Design Standard for Radiation Protection specifies requirements of the design of the ventilation systems aligned with international good practice for Radiological Protection, including the requirements for fire safety.

10.11.8 Narrative for Claim 2.2.15.4 – A4

The alignment of the HVAC design with UK RGP impacts on a wide range of topic areas including:

- Categorisation and Classification of the SMR-300 HVAC Systems.
- External Hazards Parameters.
- Radiation and Contamination Zoning.
- Fire Considerations in Relation to Containment Isolation Valves.
- I&C and Electrical.

An HVAC Working Group has been created involving topic leads from the above areas with the aim of developing a strategy for design solutions that address the requirements of all stakeholders, including both the US and UK design teams.

The output of the HVAC Working Group will be an HVAC Gap Analysis against UK RGP which will outline the further work that may be required to demonstrate risks are reduced to ALARP for a UK deployed SMR-300. Due to the nature and complexity of the work, the HVAC Gap Analysis will not be issued until the completion of GDA Step 2.

The Design Standard [44] provides the design basis for the control of airborne contamination and gaseous radiation sources, which includes requirements of ventilation systems including those which reduce the dose burden of OSW during maintenance and decommissioning. These requirements were produced to be consistent with ONR TAG-002 [83] and UK nuclear industry guidance, Ventilation Systems for Radiological Facilities, Design Guide [118]. The Design Standard [44] specifies that a series of radiological fire safety assessments must be carried out at an early stage to ensure design and operating philosophies are established wherever there is the potential for smoke and fire. The conflicting requirements of radiological safety and containment against the need to vent smoke in the event of a fire will vary according to the actual or potential radioactive inventory and zoning. Implementation of the

Design Standard in updates to the design of the RCV will ensure that UK Radiological Protection requirements of the ventilation systems will be met.

10.11.9 CAE Summary

The claim that ventilation and containment systems are designed to meet Radiological Protection requirements and minimise exposures is partially met at this stage of the SMR-300 design. The design incorporates cartridge-type HEPA filters and remote handling techniques to reduce operator exposure during filter changes. Filter housing, room layout, and shielding are being developed to support safe maintenance and minimise both internal and external doses. Airflow design ensures pressure gradients prevent the spread of airborne contamination, aligning with ALARP principles. The balance of conflicting requirements of radiological safety and containment against fire isolation will be addressed through fire safety assessments. These aspects are discussed in the Design Standard for Radiation Protection [44].

However, additional work remains to justify the HVAC design reduced risks to ALARP, particularly regarding system categorisation, radiation zoning, and ease of filter change. The current design does not fully algin with UK expectations for maintainability in areas such as equipment packaging and changeout procedures.

[REDACTED]

To address these, Holtec has established a Design Challenge and formed an HVAC Working Group with cross-disciplinary input. A HVAC Gap Analysis is underway (Mechanical Commitment C*Mech*028), formally captured in the Commitments, Assumptions and Requirements process [80], building on a high-level gap analysis conducted during GDA. This analysis will identify required design changes to achieve UK compliance and ALARP. The process and associated commitments will continue beyond GDA Step 2 to ensure all gaps are closed in the final design.

10.12 MONITORING AND ALARM SYSTEMS

Claim 2.2.15.5: The design and layout of the generic Holtec SMR-300 includes monitoring and alarm systems to alert to the presence of radiation and contamination.

Claim 2.2.15.5 has been further decomposed into three arguments to address the claim:

- Plant and systems requiring monitoring and alarms in support of normal operations have been identified (A1).
- Plant and systems requiring monitoring and alarms to alert personnel of increases in radiation or contamination levels have been identified (A2).
- Environmental monitoring systems have been identified for gaseous and aqueous discharges and direct radiation (both at point of origin and at the site boundary) (A3).

Argument 2.2.15.5 – A1: Plant and systems requiring monitoring and alarms in support of normal operations have been identified.

Argument 2.2.15.5 – A2: Plant and systems requiring monitoring and alarms to alert personnel of increase in radiation or contamination levels have been identified.

10.12.1 Evidence for Claim 2.2.15.5 – A1 and A2

Note that at this early design stage, the pieces of evidence for arguments 1 and 2 are not sufficiently detailed to differentiate between installed systems which form SSCs, such as dose rate controlled interlocks and dosimetry, and general area monitoring systems to alert personnel of deviations from normal operating conditions.

SMR-300 Design Standard for Radiation Protection [44]

The design shall provide for equipment to ensure there is adequate radiation monitoring for in plant radiation and airborne radioactivity as appropriate for normal operations, AOOs and DBAs. The standard specifies stationary alarming monitors, process system monitoring, accident monitoring, effluent monitoring and offsite monitoring.

System Specification for Monitoring and Operation [119]

The System Specification for Monitoring and Operation defines the safety and non-safety related Human System Interface (HSI) system installed in the MCR and the Remote Shutdown Facility (RSF).

System Design Description for Radiation Monitoring System [56]

The System Design Description (SDD) of Radiation Monitoring Systems defines the equipment and specification for the detectors used in the RMS. The RMS provides radiation monitoring for the four functional classifications, process monitors, airborne monitors, liquid and gaseous effluent monitors and the area radiation monitoring subsystem.

10.12.2 Narrative for Claim 2.2.15.5 – A1 and A2

The requirements for monitoring and alarm systems to alert to the presence of radiation and contamination are defined in the Design Standard for Radiation Protection [44]. This includes

the requirement for monitoring during normal operations, AOOs and DBAs. In addition, there is the need for a post-accident monitoring system to provide data on the ambient and radiological conditions inside the plant to aid planning or recovery operations to minimise exposure of the plant operating staff in the performance of necessary accident-related work activities and to evaluate the radiological conditions around the plant and assess the need for off-site action.

The Radiation Monitoring System [56] provides for operational monitoring to be undertaken across the SMR-300 plant , including:

- Reactor core: continuous neutron flux, reactor power levels.
- Containment atmosphere: monitoring for radioactive gases and particulate inside the containment building as well as detecting increases in pressure or temperature that may indicate leaks.
- Spent fuel storage: monitoring of radiation levels and water temperature in spent fuel pools.
- Ventilation system continuous monitoring of exhaust air from controlled areas for radioactive particles and gases e.g. containment air filtration exhaust; equipment floor and drainage system; radwaste, turbine and controlled area HVAC system; gaseous radwaste discharge.
- PSL: monitoring and sample extraction system for optimisation of reactor chemistry (in the CVC) and detection of potential faults. Assesses fission product concentration in coolant to identify potential fuel element failures e.g., component cooling water system, steam generator blowdown, main steam line.
- MPC monitoring. There is a requirement to monitor the MPC contents after He filling, as a final check for failed fuel as part of the fuel route.

The design also provides provisions for radiation and dose monitoring to be undertaken across work areas of the SMR-300, including:

- Workplace radiation levels: monitoring of dose rates in controlled and supervised areas during maintenance or operational activities e.g. rad chemical lab area, fuel handling area during transfers.
- Area monitoring: radiation and contamination monitoring shall be undertaken in all controlled areas. Monitoring shall be considered for some supervised and undesignated areas for assigning of doses to certain workers and demonstration of controls. Continuous radiation and contamination monitoring can provide information on trends which may indicate degradation in processes or procedures and can therefore also inform improvements to ensure doses and risks remain ALARP.
- Supply air duct: the main control room air supply duct continuously measures the concentration of radioactive materials in the air that is supplied to the main control room.

The design provides for accident and incident monitoring to be undertaken across the lifecycle of the SMR-300, including:

- Radiation alarms: real time alarms are deployed to detect sudden spikes in radiation levels, enabling prompt response. Detection methods are built into the waste management monitoring e.g. primary radwaste system liquid sample.

- Offsite monitoring: collaboration with external monitoring stations for detection of off-site releases.

Argument 2.2.15.5 – A3: Environmental monitoring systems have been identified for gaseous and aqueous discharges and direct radiation (both at point of origin and at the site boundary).

10.12.3 Evidence for Claim 2.2.15.5 – A3

PER Chapter 5 Approach to Sampling and Monitoring [35]

PER Chapter 5 presents the sampling and monitoring arrangements for radioactive waste arising from normal operation of the generic SMR-300, which aims to satisfy the information requirements in the Environment Agency's GDA guidance [120] and relevant principles in the Radioactive Substances Regulation (RSR) [121].

PER Chapter 6 Demonstration of Best Available Techniques [14]

PER Chapter 6 demonstrates compliance with the requirements set out under the Monitoring Conditions of the standard RSR environmental permit [122].

System Design Description for Radiation Monitoring System [56]

The SDD for RMS defines the equipment and specification for the detectors used in the RMS. The RMS is designed to perform plant effluent monitoring, process fluid monitoring, airborne radioactivity monitoring, and continuous indication of the radiation environment in plant areas where such information is needed.

10.12.4 Narrative for Claim 2.2.15.5 – A3

Claim 4.7, Sampling and Monitoring of PER Chapter 6 BAT [14] requires that the generic SMR-300 includes appropriate monitoring and sampling arrangements for measuring and assessing discharges, disposals and releases of radioactive waste to demonstrate compliance with the proposed limits and provide an indication of plant performance. The arguments presented in support of this claim are considered to demonstrate compliance with the requirements set out under the Monitoring Conditions of the standard RSR environmental permit [122] which include the requirements to take samples and measurements to show compliance with the RSR permit.

The Radiation Monitoring System [56] ensures that the requirements of PER Chapter 6 BAT [14] and the arrangements in PER Chapter 5 Approach to Sampling and Monitoring [35] are reflected in the design of plant effluent monitoring, process fluid monitoring, airborne radioactivity monitoring, and radiation environment monitoring in plant areas.

10.12.5 CAE Summary

The claim that the design and layout of the generic Holtec SMR-300 includes monitoring and alarm systems to alert to the presence of radiation and contamination is broadly supported at this stage of the SMR-300 design. The requirements for monitoring and alarm systems to alert to the presence of radiation and contamination during normal operations are defined in the Design Standard for Radiation Protection [44]**.** The SDD for Radiation Monitoring Systems [56] defines the equipment and specification for the detectors used in the Radiation Monitoring System.

The Radiation Monitoring System [56] ensures that the requirements of PER Chapter 6 BAT [14] and the arrangements in PER Chapter 5 Approach to Sampling and Monitoring [35] are reflected in the design of plant effluent monitoring, process fluid monitoring, airborne radioactivity monitoring, and radiation environment monitoring in plant areas. Implementation of this system should therefore ensure that impacts on MoP are ALARA and within limits.

10.13 DESIGNATION OF AREAS

Claim 2.2.15.6: Radiation zoning ensures access to and between Controlled and Supervised Areas will be suitably controlled to minimise exposures to ALARP and prevent the spread of radioactive material.

Claim 2.2.15.6 has been further decomposed into one argument to address the claim:

- Key personnel access areas and containment / layout will be mapped against these zoning schemes (A1).

Argument 2.2.15.6 – A1: Key personnel access areas and containment / layout will be mapped against these zoning schemes.

10.13.1 Evidence for Claim 2.2.15.6 – A1

SMR-300 Design Standard for Radiation Protection [44]

The SMR-300 Design Standard for Radiation Protection specifies the radiation and contamination zoning requirements that underpin the SMR-300 Dose Management Strategy [46].

SMR-300 Dose Management Strategy [46]

The Dose Management Strategy lays out the radiation and contamination zoning philosophy. It also provides initial zoning of key plant and personnel areas in the RAB and CS based on preliminary dose rate analysis and OPEX.

10.13.2 Narrative for Claim 2.2.15.6 – A1

The application of radiation and contamination zoning schemes at an early stage of the design of a nuclear facility ensures that the development of the design incorporates Radiological Protection good practice in the development of layouts, shielding containment and ventilation systems. The zoning schemes derived during the design of the facility differ from those utilised by the future site operator; however, the implementation of a zoning scheme informed byRGP at this stage should ensure that any future operators zoning schemes can be implemented more efficiently for UK deployment.

At this stage of the SMR-300 development, the design is not mature enough to rely on detailed dose rate predictions or occupancy estimates to apply a radiation zoning scheme. Instead, the zoning strategy is informed by OPEX from other PWRs, leveraging established best practices and historical data to guide the placement of critical systems and areas. This approach ensures that the zoning plan supports effective dose management to ensure exposures are ALARP, even in the absence of detailed design information. The preliminary zoning framework focuses on segregating areas by anticipated radiation levels, enabling the incorporation of shielding, access controls and operational protocols early in the design process, ensuring flexibility for future optimisation as the design progresses.

The radiation zoning scheme is presented in Appendix A and is derived from the SMR-300 zoning scheme presented in the Design Standard for Radiation Protection [44] which is based around typical schemes in the US, UK and Europe, incorporating specific UK regulatory requirements and working practices.

The contamination zoning scheme presented in Appendix B is based on existing UK practice from operational (and decommissioning) reactor sites. It is based on internal doses not exceeding 1 mSv in a calendar year. Internal dose estimates are derived using airborne contamination levels based on established Derived Air Concentration (DAC) values, which are calculated from the Annual Limit on Intake (ALI) using a dose constraint of 1 mSv per year. This approach ensures the predicted intake levels remain consistent with dose constraints. In higher contamination areas, Respiratory Protective Equipment (RPE) is worn to provide additional dose reduction, with Protection Factors selected to ensure the effective intake remains below 1 mSv. The combination of source term control, the designation of areas, the application of HoC (engineering controls, systems of work and PPE - including RPE) should ensure that no individual receives an internal dose greater than 1 mSv in a calendar year.

The initial designation of radiation areas, presented in the Dose Management Strategy [46] is based upon preliminary estimation of dose rates in the vicinity of tanks and filters of the LRW, SRW and CVC. These initial designations will inform preliminary shielding analysis and support layout optimisation.

Radiation zones will be optimised once further information becomes available. Areas will be defined and mapped based on expected radiation levels, occupancy, and operational needs (e.g., controlled, supervised, undesignated or exclusion zones).

The layout of the RAB is undergoing optimisation in terms of grouping higher radiation/ contamination areas together to minimise the need for multiple sub changes and ensuring a graduated area classification system from high to low. The current design doesn't foreclose any design modifications to change the zoning.

The current layout in the DRP (based on the SMR-160 Design Standard [96]) is not compliant with UK legislation; however, the SMR-300 obligations set out in the SMR-300 Design Standard [44] align with UK area designation legislation and guidance [60]. The assessment of space requirements for temporary containment and laydown areas to support maintenance activities has not yet been undertaken. Additionally, the adequacy of the central changeroom to accommodate the anticipated number of personnel and associated equipment is undergoing evaluation. Consideration has been formally given for refuelling outages in Outage Strategy for SMR-300 [123] - specifically in scheduling of tasks, major lifts and equipment movements between containment and the RAB. Informally, the NQA-1 design process considers EIMT inherently to the production of GA drawings (see Chapter A2 & A4, [8] [12]).

10.13.3 CAE Summary

The claim that radiation zoning ensures access to and between Controlled and Supervised Areas will be suitably controlled to minimise exposures to ALARP and prevent the spread of radioactive material. Is largely substantiated at this stage of the SMR-300 design. Zoning schemes, consistent with UK RGP, have been derived to enable development of the design of the SMR-300. These schemes identify the need for barrier controls between zones and indicate the level of access control required to enter.

The radiation zoning scheme for implementation of the design of the SMR-300 in the UK, presented in Appendix A, is derived from the SMR-300 zoning scheme presented in the Design Standard for Radiation Protection [44] incorporating specific UK regulatory requirements and working practices. Preliminary zoning of plant in the RAB and containment

has been completed to inform shielding and access arrangements in development of the design.

The contamination zoning scheme, presented in Appendix B, is based on existing UK practice. This scheme will be applied to the design of containment and ventilation systems, and, together with the output of fault analysis, will inform the update to the layout of the RAB. The designation of areas and the application of HoC should ensure that no individual receives an internal dose greater than 1 mSv in a calendar year.

However, gaps remain due to the immaturity of the design, which currently lacks detailed dose rate and occupancy data. As a result, zoning is based on operational experience from other PWRs rather than site-specific analysis. Additionally, space planning for change rooms and temporary containment areas is still under assessment.

10.14 OUT-OF-CORE CRITICALITY SAFETY

Claim 2.2.15.7: Criticality controls are implemented to ensure that criticality risks are reduced to ALARP.

Claim 2.2.15.7 has been further decomposed into one argument to address the claim:

- Controls are in place for the management of new and spent fuel through the fuel lifecycle to ensure the risk of a criticality is ALARP (A1).

Argument 2.2.15.7 – A1: Controls are in place for the management of new and spent fuel through the fuel lifecycle to ensure the risk of a criticality is ALARP.

Out-of-core criticality safety concerns the prevention of, or minimisation of the probability and severity of a criticality incident involving new or spent fuel outside of the RPV. This relates to Regulation 9 of IRR17 [15] regarding restriction of exposure, and Regulation 8 regarding radiation risk assessment. Criticality control inside the RPV is addressed in Part B Chapter 2 [9].

10.14.1 Evidence for Claim 2.2.15.7 – A1

SMR-160 Fuel Handling and Storage [124]

This report provides a description and the design basis for maintaining the fuel assemblies subcritical while in the New Fuel Vault (NFV), in the SFP, and in the Multi-Purpose Canister (MPC) that is used during refuelling operations.

Specification for SMR-160 New Fuel Storage Racks [125]

Defines the criticality safety requirements for the New Fuel Storage Racks.

Specification for SMR-160 Spent Fuel Storage Racks [126]

Defines the criticality safety requirements for the Spent Fuel Storage Racks.

Chapter B24 Fuel Transport and Storage [28]

The Fuel Transport and Storage operations within the scope of this chapter include:

- [REDACTED]

10.14.2 Narrative for Claim 2.2.15.7 – A1

The Fuel Transport and Storage design includes operations where arrays of New Fuel Assemblies (NFA) or Spent Fuel Assemblies (SFA) are brought together in fuel storage racks in the SFP or in an MPC for interim storage. Controls are in place to ensure these arrays can be shown to be criticality safe during normal operations, abnormal operating occurrences and accident conditions. Criticality is controlled through the fuel lifecycle by geometric spacing of the NFA and SFA, the presence of neutron absorbing materials and managing the inventory of nuclear material present in the array. For Criticality Safety Assessments the design basis effective neutron multiplication factor is limited to a Keff which does not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

10.14.2.1 New Fuel

[REDACTED]

10.14.2.2 Spent Fuel

[REDACTED]

10.14.3 Criticality Safety Assessments

The criticality safety assessments will be developed by a future site operator at the site-specific stage. Criticality calculations are performed using proven and validated computer codes.

10.14.4 CAE Summary

The claim that out of core criticality controls are implemented to ensure that criticality risks are reduced to ALARP is fundamentally supported at this stage of the SMR-300 design. The SMR-300 design includes operations where arrays of NFA or SFA are brought together in fuel storage racks in the SFP or in MPCs for interim storage. Criticality is controlled through the fuel lifecycle by geometric spacing of the NFA and SFA, the presence of neutron absorbing materials and managing the inventory of nuclear material present in the array. These are passive engineered safety measures consistent with the hierarchy of protection in SAPS [61] Principle ECR.1 and is consistent with the Double Contingency Approach (DCA) in Principle ECR.2. For Criticality Safety Assessments the design basis effective neutron multiplication factor is limited to a Keff which does not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

10.15 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the Radiological Protection chapter and how this chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 [29] 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 RGP.
  - Evaluation of Risk and Demonstration Against Risk Targets.
  - Options Considered to Reduce Risk.
- GDA Commitments.
- Conclusion.

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

10.15.1 Technical Summary

Part B Chapter 10 aims to demonstrate the following Level 3 claims to a maturity appropriate for a PSR:

Claim 2.1.1: Radiological Protection requirements are identified such that effective doses to workers and public during normal operations are below legal limits, defined justified dose targets and constraints and are As Low As Reasonably Practicable.

Claim 2.2.15: SSCs are designed to meet Radiological Protection requirements and minimise exposures.

This has been achieved by summarising the evidence available for each of the Level 4 claims, which collectively show that the Level 3 claims are met. The Level 4 claims are discussed below.

Claim 2.1.1.1: The generic Holtec SMR-300 is compliant with relevant Radiological Protection legislative requirements.

The generic Holtec SMR-300 is shown to be broadly compliant with UK Radiological Protection legislative requirements. This is demonstrated by the following:

- A review of relevant UK legislation against US legislation to ensure all UK requirements are at least met [43].
- A review against IRR17 [15], its ACoP [60] requirements and ONR SAPs RP1-7 [61].

Any gaps between US and UK requirements have, in the first instance, been addressed in the Dose Management Strategy [46] or the Design Standard for Radiation Protection [44]. Any other gaps which would affect the design and operation of the SMR-300 have been raised as risks and will be managed as part of the Risk Management Plan [79].

Claim 2.1.1.2: Doses associated with the generic SMR-300 are below legal limits and are demonstrated to be ALARP.

The Dose Management Strategy [46] and the Design Standard for Radiation Protection [44] define the dose constraints and limits. Dose constraints have been derived based on analysis of dose performance from the operation of PWRs in the UK, US and Europe. The constraints set aim to be aspirational but achievable, cognisant of the differences between an SMR and traditional PWR reactors. Methods for assessing doses to OSW and MoP during normal operations are presented in the Dose Management Strategy [46]. These documents also define the process to be used to undertake NODA and ALARP studies. NODA and ALARP studies have not been undertaken at PSR V1 due to limited design, dose rate and occupancy data. However, RGP will be utilised when such studies are undertaken to ensure radiological hazards and doses are reduced to ALARP. UK legislation compliant radiation and containment zoning schemes have been developed which are presented in Appendix A and Appendix B, respectively. NODA will be carried out as part of business as usual as the design develops to support design decisions and ensure doses and risks are ALARP.

Claim 2.1.1.3: The safety hierarchy of controls will be applied to ensure radiological hazards will be reduced to ALARP.

The Dose Management Strategy [46] describes the structured, hierarchical approach to the management of risks whereby risks are reduced using the ERICPD approach. This is aligned with the IRR17 [60], HoC for managing radiation risks.

The primary aim is to eliminate the radiation hazard by choice of materials, chemistry control and passive safety systems that require no operator action.

Where elimination is not possible, or radiological hazards persist, HoC emphasises the importance of implementing engineering controls such as shielding, interlocks and containment, supported by administrative controls such as work planning and procedural access restrictions and finally PPE.

Claim 2.2.15.1: Sources of residual radiological risk are appropriately characterised to inform the design of SSCs.

Source terms for gamma and neutron emissions, decay heat and radiological inventory have been derived for the reactor and SFP for use in shielding, radiological consequence, radioactive waste management and effluent discharge assessments. These source terms will be used to inform the design of SSCs. Further source terms are under development, and all source terms will undergo review and revision as the design develops.

Claim 2.2.15.2: The generic Holtec SMR-300 shielding is designed and substantiated to minimise exposures for all plant areas and operation stages, including waste package transport.

The Shielding Basis of Design [47] presents preliminary dose rate calculations based on defined source terms, preliminary shielding thicknesses and associated material properties for the reactor core and SFP. The report determines the maximum dose rate outside containment as a result of the reactor core at full power to be far below the limit of 0.5 μSv/h for R0

undesignated areas as defined in Appendix A. The dose rates from the SFP contents are well below background and will not contribute significantly to total dose rates inside or outside of the containment. This preliminary analysis gives confidence that the proposed shielding design is capable of achieving dose targets once all other sources are taken into consideration. Optimisation of the shielding design will be undertaken in conjunction with NODA in order to demonstrate that normal operation dose uptake is ALARP for all OSW and MoP. The layout of the RAB has taken into consideration OPEX for shielding thicknesses across US reactor fleet in determining the preliminary wall thicknesses.

Claim 2.2.15.3: Materials selection minimises the generation of neutron activation products in SSCs.

The SMR-300 has been designed from the outset to ensure that materials are not susceptible to neutron activation, as detailed in the Minimalization Strategy [110]. The Dose Management Strategy [46] presents the SMR-300 strategy for material selection. Materials containing cobalt, silver and antimony in components near the reactor core which give rise to out of core dose rates are minimised e.g. there is no use of Stellite in lower core internals. The shielding philosophy for the SMR-300 includes selection of shielding materials designed to reduce neutron activation of the shielding, surrounding structural and functional components. Consideration will be given to numerous other chemical elements that could become activated and pose a significant dose or waste handling / disposal-related hazard.

Claim 2.2.15.4: Ventilation and containment systems are designed to meet Radiological Protection requirements and minimise exposures.

Holtec have committed to analysing the differences between US and UK nuclear ventilation standards to ensure that the design of the SMR-300 ventilation systems will be developed taking cognisance of RGP with respect to containment of radioactive materials, zoning of staffed areas and the minimisation of the radiation and contamination hazards to the operator during maintenance and decommissioning of RCV components including filters and ducts. Radiological Protection aspects of nuclear ventilation design are discussed in the SMR-300 Design Standard for Radiation Protection [57].

Claim 2.2.15.5: The design and layout of the generic Holtec SMR-300 includes monitoring and alarm systems to alert to the presence of radiation and contamination.

The requirements for monitoring and alarm systems to alert to the presence of radiation and contamination during normal operations are defined in the Design Standard for Radiation Protection [44]**.** The SDD of Radiation Monitoring Systems [56] defines the equipment and the specification for the detectors used in the Radiation Monitoring System.

The Radiation Monitoring System [56] ensures that the requirements of PER Chapter 6 BAT [14] and the arrangements in PER Chapter 5 Approach to Sampling and Monitoring [35] are reflected in the design of plant effluent monitoring, process fluid monitoring, airborne radioactivity monitoring, and radiation environment monitoring in plant areas.

Claim 2.2.15.6: Radiation zoning ensures access to and between Controlled and Supervised Areas will be suitably controlled to minimise exposures to ALARP and prevent the spread of radioactive material.

The radiation zoning scheme (see Appendix A) derived to support the design of the SMR-300 is based upon the SMR-300 zoning scheme presented in the SMR-300 Design Standard [44], incorporating specific UK regulatory requirements and working practices.

The contamination zoning scheme presented in Appendix B is based on existing UK practice from operational (and decommissioning) reactor sites. The designation of areas and the application of HoC should ensure that no individual receives an internal dose greater than 1 mSv in a calendar year.

Claim 2.2.15.7: Criticality controls are implemented to ensure that criticality risks are reduced to ALARP.

The SMR-300 design includes operations where arrays of NFA or SFA are brought together in fuel storage racks in the SFP or in MPCs for interim storage. Criticality is controlled through the fuel lifecycle by geometric spacing of the NFA and SFA, the presence of neutron absorbing materials and managing the inventory of nuclear material present in the array. For Criticality Safety Assessments the design basis effective neutron multiplication factor is limited to a Keff which does not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

10.15.2 ALARP Summary

10.15.2.1 Demonstration of the use of Relevant Good Practice

The generic Holtec SMR-300 is broadly compliant with UK Radiological Protection legislative requirements and guidance. This is demonstrated by the following:

- A review of relevant UK legislation against US legislation to ensure all UK requirements are met or exceeded [43].
- A review against IRR17 [60], its ACoP requirements and SAPs RP1-7 [61].

Any significant regulatory gaps between US and UK requirements have been addressed in the Design Standard for Radiation Protection [44] where UK requirements align with international BSS [93] [94]. UK specific requirements were captured within the Dose Management Strategy [46]. Other gaps, including those that could not be fully met through the inclusion in the Design Standard, which would affect the design and operation of the SMR-300 have been raised as risks and will be managed through the Risk Management Plan as part of normal business [79].

The most significant differences between the DRP based on US expectations and UK requirements relate to personnel access to the reactor containment at power; the ventilation design for containment; the provision of shielding, and the design of change rooms and barriers for accessing controlled areas. These are discussed below.

10.15.2.1.1 Ventilation design for containment

The current SMR-300 HVAC design does not fully meet UK regulatory standards due to differences in requirements for controlled and uncontrolled areas and nuclear ventilation standards. The DAC values at which to designate controlled areas in the UK are lower than in the US. In addition to the difference in requirements for radiation zoning the annual dose limit is 2.5 times higher in US resulting in the US DAC being 2.5 times higher. A Design Challenge [81] has been raised, in part, to address this. The Design Standard for Radiation Protection

[44] defines a Contamination Zoning System which is based on existing UK practice from operational (and decommissioning) reactor sites. Further work is required to confirm that the zoning and ventilation requirements proposed in the Design Standard for Radiation Protection [44] meets UK legislative requirements and RGP. A Commitment is raised in sub-chapter 10.15.3 to address this issue.

10.15.2.1.2 Design of Controlled Areas in the RAB

The RAB layout is still undergoing design optimisation and is subject to update as the design matures, and inherent risks are better understood as addressed in Chapter A4 (4.8.2) [22]. Additionally, EIMT is being considered in line with sub-chapter 9.6 of Chapter B9 [23].

[REDACTED]

These considerations will be incorporated into the ongoing design development to ensure appropriate provision and compliance with operational needs.

Application of the requirements stipulated in the SMR-300 Design Standard [44] regarding area designation, change room requirements and access controls will achieve the UK and international regulatory standards; however, the current layout of the RAB (in DRP 1.1 [78]), is not fully compliant with the obligations in the SMR-300 standard, as it is designed against the SMR-160 Standard which was based on 10 CFR 20 [63] and associated NRC regulatory guidance [67].

10.15.2.1.3 Entry at Power to the Containment Structure

Personnel may infrequently be required to enter reactor containment at power to investigate faults and carry out repairs without necessitating a full reactor shutdown. This will be agreed on a case-by-case basis. Preliminary shielding assessment in the Shielding Basis of Design [47] shows that dose rates inside the CS when the reactor is at power are in the range mSv to kSv. Therefore, access at full power cannot be deemed to meet Regulation 9(1) or Regulation 12(1) [60] under the DRP.

The containment and reactive actions identified to allow entry at power consists of identification of scenarios which would require personnel access while the reactor is on load, followed by ALARP optioneering to identify alternative methods to prevent or mitigate the risk which will ensure that doses to employees are ALARP.

10.15.2.1.4 Shielding

Shielding design is not fully matured, and provisional shielding arrangements in the design are based on US OPEX and area designations. The SMR-300 Design Standard for Radiation Protection [44], and associated Equipment and Piping Guidelines [109] more closely align with UK and international regulatory expectations, therefore any updates to the shielding design should more closely align with UK standards.

As shielding is currently based on US OPEX and shielding requirements have not yet been fully assessed, layout, civil and structural engineering are likely to be impacted as the requirements in the Design Standard for Radiation Protection [44] are incorporated. Impacts and changes to the layout will be mitigated/managed by the design configuration control processes, as outlined in Chapter A4 (4.8.2) [22].

10.15.2.2 Evaluation of Risk and Demonstration Against Risk Targets

The numerical targets against which the demonstration of ALARP is considered can be found in Part A Chapter 2 [8].

The consideration of normal operations doses to workers and the public contributes to the design development of the generic SMR-300, supporting existing decisions where appropriate or informing improvements to the design.

The SMR-300 has been designed to demonstrate that doses are ALARA [44] and below the defined and justified dose targets and limits. Normal operations cover Level 1 and 2 of the defence-in-depth expected for demonstrating the safety of a nuclear power plant, which are:

- Level 1 – the prevention of abnormal operation and failure by design.
- Level 2 – the prevention and control of abnormal operation and detection of failures.

By looking at risks from radiation exposure directly, the contributions to individual dose from lifetime (stochastic) factors (i.e., the build-up of everyday exposure) make an implied determination of radiological risk and contribute to the demonstration of ALARP. Hence, the risk from normal operations covers the effective dose uptake during the following:

- Reactor operations:
- Start-up.
- Shut-down.
- Full-power.
- Abnormal.
- Refuelling.
- Examination, Maintenance, Inspection and Testing (EIMT).
- Operations outside of containment:
- Radioactive source handling operations.
- Fuel movements.

For normal operations, the dose targets given in Target 1-3 of Part A Chapter 2 [8] are derived from the Ionising Radiation Regulations 2017 (IRR17) [15] for OSW and MoP. The fundamental principles of ALARP in radiological protection in the UK are documented in IRR17, (especially Regulation 9), TAGs (NS-TAST-GD-038 [85] and NS-TAST-GD-043 [82]), TIGs, SAPs and ACOP [127] . Targets 1-3 are presented in Table 7 and in PSR Part A Chapter 5 [29].

SSCs associated with Radiological Protection will contribute to the demonstration of ALARP by fulfilling safety functions for normal operations (e.g. reduction of normal operation dose uptake by provision of shielding and optimisation of time and distance) and thereby contributing to achieving Targets 1-3.

As accident doses are assessed and optimised within Part B Chapter 14 [6], this document does not directly contribute to the assessment of accident risk. However, radiation source terms are derived as part of Part B Chapter 10.

The evaluation of the normal operation risks against Targets 1-3 is summarised in Part A Chapter 5 [29].

10.15.2.3 Options Considered to Reduce Risk

The process for the assessment of risk reduction options is presented in the Design Management Process [128]. Part A Chapter 5 [29] considers the holistic risk-reduction ALARP process for the generic SMR-300.

At this stage of the SMR-300 design, only one Design Challenge with an impact on Radiological Protection has been identified relating to HVAC Architecture, Design Codes and Design Basis [81] and this is identified as a GDA Commitment in sub-chapter 10.15.3. Risk reduction options will be identified and ALARP studies undertaken to optimise the risk.

Sub-chapter 10.15.2.1.3 provides discussion on the containment and reactive actions identified for reconciling the entry at power issue which included ALARP optioneering to identify alternative methods to prevent or mitigate the risk of worker exposure such that they are ALARP.

10.15.3 GDA Commitments

GDA Commitments which relate to this chapter have been formally captured in the CAR Register [80]. The following GDA Commitment identified for Part B Chapter 19 [30] applies to Part B Chapter 10:

C_Mech_028: A high level gap analysis has determined that UK RGP differs from the US approach to designing HVAC SSCs in several areas of mechanical design, including requirements related to Fire Protection, Radiation Protection, EC&I, External Hazards, and SSC classification. A Commitment is raised to review and incorporate UK RGP into the design of HVAC SSCs in the UK SMR-300 where it is practicable. Target for resolution: Issue of UK Pre-Construction SSEC.

10.15.4 Conclusion

This chapter summarises the approach to Radiological Protection for the generic SMR-300 in the UK. It identifies the claims and arguments that will form the basis of the safety case for the Radiological Protection 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.

The key aspects of Radiological Protection are the HoC including engineered controls (access controls, containment and shielding) designed to optimise doses, and production of a NODA that demonstrates dose constraints and targets can be achieved and dose uptake is ALARP.

The Shielding Basis of Design [47] presents preliminary dose rate calculations for the reactor core at power and a loaded SFP, based on defined source terms, preliminary shielding thicknesses and associated material properties. The report determines the maximum dose rate outside containment from the reactor core at power to be [REDACTED] µSv/h which is below the limit of 0.5 µSv/h for R0 undesignated areas as defined in Appendix A. The dose rates from the SFP contents are well below background and will not contribute significantly to total dose rates inside or outside of the containment. This preliminary analysis gives confidence that the proposed shielding design in the CS/CES will enable the dose targets outside of containment to be achieved, for the reactor. Optimisation of the shielding design will be undertaken in order to demonstrate that normal operation dose uptake is ALARP.

The Dose Management Strategy [46] and the SMR-300 Design Standard for Radiation Protection [44] define the dose constraints and limits, and methods for assessing doses to OSW and MoP during normal operations. They also define the process to be used to undertake NODA and ALARP studies. NODA and ALARP studies have not been undertaken at PSR V1 due to limited design, dose rate and occupancy data. However, assessment against RGP will be required when such studies are undertaken to ensure radiological hazards and doses are reduced to ALARP.

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Appendix A External Radiation Controlled Area Zones

Area Designation	Indicative Dose Rates µSv/h	Max Occupancy per Year (1mSv)	Description
Undesignated R0 (white)	≤ 0.5	Unrestricted	Areas of site that can be occupationally occupied without exceeding annual dose limits for members of the public of 1 mSv/y. Typical areas include offices and ancillary buildings.
Radiation Supervised Area R1 (green)	0.5 to 2.5	Unrestricted	Areas of plant that can be occupationally occupied without exceeding personnel radiation monitoring requirements (>1 mSv) or >1/10 of any relevant dose limit, or where it is necessary to keep conditions under review to determine whether the area needs to be a Controlled Area. Areas designated R1 are generally high occupancy areas with the potential for slightly elevated or transient doses.
Radiation Controlled Area R2 (yellow)	2.5 to 25	40-400 h	Areas of plant that can be occupationally occupied without exceeding annual occupational dose limit (20 mSv). Potential for personnel to receive a dose greater than 6 mSv/y. Personnel monitoring required. Annual dose constraint could be exceeded in a month. Areas designated R2 may have frequent, short-term occupancy or occasional longer-term occupancy, access will be via barrier control.
Radiation Controlled Area – limited occupancy R3 (orange)	25 to 50	20-40 h	Areas of plant that require limited access to ensure compliance with the annual occupational dose limit. Annual dose constraint could be exceeded in a week. Access to R3 areas will be strictly controlled. Occupancy will be infrequent, for short time periods under written systems of work via barrier control.
Radiation Area R4 (red)	50 to 1000	Routine access restricted (1-20h)	An area, subject to high dose rates (permanent, temporary, or transient), where the annual dose constraint could be exceeded in a day. Access to R4 areas will be restricted, under a written system of work via barrier control and strict dose uptake controls to limit exposures. Active dosimetry required.
Restricted Access High Radiation Area R5 (black)	>1000	Routine access prohibited	An area subject to very high dose rates (permanent, temporary, or transient). Access to R5 areas will be strictly prohibited unless for safety critical purposes. Areas to be locked off with access under strict controls, for EIMT or emergency purposes only.

Appendix B Contamination Controlled Area Zoning

Area Designation	Indicative Activity Levels	Description
Non designated C0	N/A	Areas where radioactive contamination is not anticipated to occur, for example offices and areas of the RAB before the main change room.
Contamination Supervised Area C1	Less than C2 lower limits	An area where radiological conditions will be kept under review; for example, the clean side of a sub-change room. Typically, areas with the potential for contamination to spread.
Contamination Controlled Area C2	Alpha emitters > 0.4 Bq/cm^2 Radionuclides not otherwise specified (including ^{3}H) > 4 Bq/cm^2	An area where airborne contamination is not expected to be present, persons are unlikely to receive an internal exposure of 1 mSv in a calendar year. To prevent the spread of surface contamination beyond the area, barriers, PPE and housekeeping will be utilised.
Contamination Controlled Area C3	Alpha N/A Beta / gamma > 10 Bq/m^3	Where alpha emitting radionuclides may be disregarded. Use of barriers/containment, PPE and ventilation to prevent the spread of airborne contamination beyond the area. Access will be under a written system of work.
	Alpha > 0.01 Bq/m^3 Beta/gamma > 2 Bq/m^3	Where alpha and beta / gamma emitting radionuclides are present. Additional PPE, including RPE may be identified in radiation risk assessment. Access will be under a written system of work.
	Alpha N/A Beta/gamma > 1 10^4 Bq/m^3	Where tritium is present, and alpha and other beta / gamma emitting radionuclides may be disregarded. Additional PPE, including RPE may be identified in radiation risk assessment. Access will be under a written system of work.
Contamination Controlled Area C4	>100 x the lower level for C3	Areas where access is generally prohibited, for example inside vent plant. As for C3, type of RPE to be used to be identified in radiation risk assessment.

Appendix C Safety Functions for Radiological Protection SSCs

Plant-Level Safety Function or Operational Function	Plant-level System or Group	SSC	Performance Requirement
Maintain Reactor Coolant Pressure and Containment Boundary Integrity Charging Isolation. Degasification of the Reactor Coolant System.	Chemistry Control	CVC (Charging)	SDD for Chemical and Volume Control System [50]
Maintain the Reactor Coolant Pressure Boundary Integrity and maintain the Containment Boundary Integrity. Monitor chemistry of Nuclear Island systems. Obtain representative sample. Provide protection against exposure and contamination during collection.	Chemistry Control	Primary Sampling System	SDD for Primary Sampling System [51]
Provide capability to isolate main control room. Provide adequate protection to the main control room personnel to ensure radiation exposure is within acceptable limit. Provide habitability for MCR during all modes of operation.	Non- Radiological Controlled Area HVAC (NRV)	Control Room Normal Ventilation System (CRV) Air Filtration Unit as part of the NRV	Position Paper on HVAC Systems [129]
Containment isolation to maintain the containment boundary and vacuum relief to maintain the containment integrity.	Containment Building Ventilation System	CBV Cooling Fan	SDD for Containment Ventilation System [52]
Provide ventilation and temperature control to permit personnel access and to control the concentration of airborne radioactive material during normal operation.	HVAC	RCV	Position Paper on HVAC Systems [129]
Maintain the containment boundary integrity. Receive and collect radioactive waste gases generated during normal plant operation. Condition waste gases prior to treatment Retain off-gases for radioactive decay.	Radioactive Waste Systems	Gaseous Radwaste System	SDD for Gaseous Radwaste System [53]
Maintain the containment boundary integrity. Control, collect, process, handle, store, and dispose of liquid radioactive waste. Process and dispose of steam generator blowdown if significant radioactivity is detected.	Radioactive Waste Systems	Liquid Radwaste System	SDD for Liquid Radwaste System [54]

Plant-Level Safety Function or Operational Function	Plant-level System or Group	SSC	Performance Requirement
Collect, hold, process, sample, package, and store wet solid radioactive wastes. Collect, segregate, sample, package, and store dry solid radioactive wastes. Collect, sample, package, and storage miscellaneous solid radioactive wastes. Provide the means to return excess radioactive liquid waste to the LRW.	Radioactive Waste Systems	Solid Radwaste System	SDD for Solid Radwaste System [55]
Determine concentrations of radioactive materials in plant fluid systems. Airborne monitors provide operators with information on concentrations of radioactivity at various points in the ventilation system. The area radiation monitoring sub system provides radiation measurements at fixed locations.	Control & Instrumentation	Radiation Monitoring System	SDD for Radiation Monitoring System [56]
Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.	Containment Structure Containment Enclosure Structure Reactor Auxiliary Building Intermediate Building Interim Spent Fuel Storage Installation	Shielding Systems	Design Specification for Containment Structure [58] Design Specification for Containment Enclosure Structure [106] Design Specification for Reactor Auxiliary Building [59].

Appendix D PSR Part B Chapter 10 CAE Route Map

Table 11: PSR Part B Chapter 10 CAE Route Map

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