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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 UK to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment - as defined in Holtec SMR GDA Preliminary Safety Report (PSR) PART A Chapter 1 Introduction [1].

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

Within this Chapter As Low As Reasonably Achievable (ALARA) is used on occasions where the context is different. When talking about ALARA in the context of the US design this is referring to ALARA from a US perspective with regards to Radiological Safety practices and the minimisation of doses to On Site Workers (OSWs) which can be considered broadly similar to ALARP in the UK. When ALARA is used in the UK context this is a result of the Best Available Technique (BAT) to ensure exposures to Members of the Public (MoP) and the environment are minimised.

PART B Chapter 10 of the PSR presents the Claims, Arguments and intended Evidence (CAE) for the radiological protection aspects of the generic SMR-300.

10.1.1 Purpose and Scope

Key outputs of this Chapter are to ensure the design:

1. Is compliant and meets with United Kingdom (UK) regulatory requirements and Relevant Good Practice (RGP).
2. Ensures that UK dose limits will not be exceeded.
3. Meets the requirements of ALARP.
4. Is at least comparable in terms of radiological doses to similar designs throughout the world.

The Overarching SSEC Claims are presented in, Holtec SMR GDA PSR Part A Chapter 3 Claims, Arguments and Evidence [2]. This chapter 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 are developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

As set out in Part A Chapter 3 [2], Claim 2.1 is further decomposed across several disciplines which are responsible for assessment of the plant operation. This chapter presents the

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radiological protection aspects of the generic SMR-300 and therefore directly supports a claim focused on the identification of radiological protection requirements for Normal Operations (NOs), such that they are ALARP, Claim 2.1.1.

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.

Further discussion on how the Level 3 claim is broken down to Level 4 claims and then how they are demonstrated is provided in subchapter 10.3.

As set out in Part A Chapter 3 [2], Claim 2.2 is further decomposed across several disciplines to consider the critical safety functions. A Level 3 claim is related to Critical Safety Function 3, Claim 2.2.3. Claim 2.2.3 is brought together in Holtec SMR GDA PSR PART B Chapter 1 Description of the Reactor Coolant System and Engineered Safety Features [3].

Claim 2.2.3: Adequate provision for the control of radiation exposure and control of release of radioactive material is incorporated into the design.

This chapter supports this Level 3 claim and is decomposed into a Level 4 claim relating to the design of the Structures Systems and Components (SSCs) for radiological protection, Claim 2.2.3.4.

Claim 2.2.3.4: SSCs are designed to meet Radiological Protection requirements and minimise exposures.

Further discussion on how the Level 4 claim is broken down to Level 5 claims and then how they are demonstrated is provided in subchapter 10.3.

This chapter covers the radiological protection aspects for NOs, which includes all normal operating states – start-up, shut-down, full-power operation, Examination, Inspection, Maintenance and Testing (EIMT) refuelling and abnormal operations. Outside of the reactor, NOs 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:

- Containment Enclosure Structure (CES) including Containment Structure (CS) and Associated Systems.
- Reactor Auxiliary Building (RAB) and Associated Systems.
- Intermediate Building (IB).
- Radioactive Waste Building (RWB) and Associated Systems.
- Interim Spent Fuel Storage Installation (ISFSI).

This chapter covers the codes, standards and methodologies for radiological protection (subchapter 10.4), radiological protection requirements (subchapter 10.5), the design of SSCs for radiological protection (subchapter 10.6), the assessment of radiological consequences and requirements (subchapter 10.7), and finally the contribution to ALARP (subchapter 10.8).

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There are no novel aspects to radiological protection within the SMR-300 design and its application in the UK.

Aspects excluded from the scope of this Chapter are:

- Radiological impact on MoP and the environment from all discharge routes, which is covered in, Holtec SMR GDA Preliminary Environmental Report (PER) Chapter 3 Radiological Impact Assessment (RIA) [4].
- Non-radiological environmental impact from NOs, which is covered in, Holtec SMR GDA PER Chapter 4 Conventional Impact Assessment [5].
- Radiological consequence analysis for fault conditions and the minimisation of accident source terms. These are addressed in, Holtec SMR GDA PSR Part B Chapter 14 Design Basis Accident Analysis [6]
- Decommissioning and the provision of information, radiation protection and the design for decommissioning are covered in, Holtec SMR GDA PSR Part B Chapter 26 Decommissioning Approach' [7].

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in Holtec SMR GDA PSR Part A Chapter 2 'General Design Aspects and Site Characteristics' [8]

10.1.2 Assumptions

No assumptions are identified in this revision.

10.1.3 Interfaces with other SSEC Chapters

This radiological protection topic interfaces with many other topics in the PSR and the PER.

PSR PART A Chapter 2 'General Design and Site Characteristics' [8] presents an overview of the design evolution, the generic design of the SMR-300 presented for the GDA, the fundamental design and safety principles and the reference design for the generic SMR-300. Chapter A2 [8] 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, RIA, [4].

PSR PART B Chapter 1 [3], Holtec SMR GDA PSR Part B Chapter 2 Reactor Fuel and Core [9] and, Holtec SMR GDA PSR 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.

Holtec SMR GDA PSR Part B Chapter 4 Control and Instrumentation Systems (C&I) [11] describes the key functions of C&I systems, their design principles, and their significance in maintaining safe and reliable operation. Radiological protection SSCs (e.g., instrumentation for radiation monitoring) that are C&I-based will be substantiated within the C&I topic.

Holtec SMR-300 GDA PSR PART B Chapter 11 'Environmental Protection' [12] and Holtec SMR GDA PSR Part B Chapter 13 'Radioactive Waste Management' [13] 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 [13] also describes the design of the radwaste systems, effluent

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discharge systems e.g., filter/resin changes to be optimised with worker dose and discharges. The approach to and application of BAT (and therefore the UK definition of the ALARA principle as applies to all aspects of the management of radioactive substances and wastes, including their disposal) works alongside ALARP to provide a holistic optimised solution.

Holtec SMR GDA PSR Part B Chapter 20 Civil Engineering [14] describes the building containment and structures of the NI, including the CES, CS, RAB, RWB and ISFSI. Shielding is designed to provide protection to potential exposed groups.

PSR PART B Chapter 14 'Design Basis Accident Analysis' [6] presents the deterministic 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 Chapter A2 [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 PSR Part B Chapter 21 External Hazards [15] and Holtec SMR GDA PSR Part B Chapter 22 Internal Hazards [16].

Holtec SMR GDA PSR Part B Chapter 15 BDBA, Severe Accidents Analysis and Emergency Preparedness [17] 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.

Holtec SMR GDA PSR Part B Chapter 16 Probabilistic Safety Assessment [18] 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.

Holtec SMR GDA PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [19], Holtec SMR GDA PSR Part B Chapter 9 Conduct of Operations [20], Holtec SMR GDA PSR Part B Chapter 25 Construction and Commissioning [21], Holtec SMR GDA PSR Part B Chapter 12 Control of Non-Radiological Hazards [22], Holtec SMR GDA PSR Part B Chapter 17 Human Factors [23] and Chapter B26 '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 [20] also includes the process for development of EIMT schedules and the description of reactor operating states. PART B Chapter 17 [23] 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.

Holtec SMR GDA PSR Part B Chapter 23 Reactor Chemistry [24] describes the source term from the reactor primary and secondary coolant, material selection and optimisation to reduce doses So Far As Is Reasonably Practicable (SFAIRP).

Holtec SMR GDA PSR Part B Chapter 24 Fuel Transport and Storage [25] describes the movement, handling and storage of new and used fuel outside of containment.

Holtec SMR GDA PSR Part A Chapter 5 Summary of ALARP [26] is intended as a summary of the SSEC and naturally links to all SSEC reports and their respective chapters.

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10.2 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 main buildings description. This chapter refers to the following SSCs, which have radiological protection safety functions:

- Chemical and Volume Control (CVC) System.
- Primary Sampling System (PSL).
- Containment Building Ventilation (CBV).
- Control Room Normal Ventilation System (CRNVS).
- Heating, Ventilation and Air Conditioning system (HVAC).
- Gaseous Radwaste System (GRW).
- Liquid Radwaste System (LRW).
- Solid Radwaste System (SRW).
- Radiation Monitoring System (RMS).
- Shielding.

Further description of these systems can be found in the subchapters below.

Containment isolation valves for the above systems are adopted, where reasonably practicable, to support the maintaining of containment integrity.

10.2.1 Chemical and Volume Control (CVC) System

The CVC System controls the reactor coolant system inventory to ensure radiological source terms in the Reactor Coolant System (RCS) are minimised during operation among other functions.

10.2.2 Primary Sampling System (PSL)

The PSL delivers liquid and gaseous samples from various points in containment and the Reactor Auxiliary Building (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 RC are kept under constant review.

10.2.3 Containment Building Ventilation (CBV)

The CBV is designed to control the containment temperature within a suitable range, prevent damage to the containment structure or equipment inside containment, and to allow conditions for personnel to perform work, when accessible. The system is designed to control airborne contamination to within acceptable limits. See subchapter 10.6.4 for discussion of potential contamination zoning schemes.

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.

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10.2.4 Control Room Normal Ventilation System (CRNVS)

The CRNVS is designed to provide a reliable source of heating, ventilation, and cooling to areas served when ac power is available. Redundant safety-related radiation monitor 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 gaseous radioactivity concentrations and isolate the Control Room Emergency Zone (CREZ) on high-high particulate or iodine radioactivity concentrations.

10.2.5 Heating, Ventilation and Air Conditioning (HVAC)

Filtration of incoming and exhaust 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 contamination and gaseous radiation sources present in the operation of the plant. These ventilation systems include the Radiologically Controlled Area HVAC (RCV), CBV, and the Radioactive Waste Building HVAC (RBV). 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.

10.2.6 Gaseous Radwaste System (GRW)

Although the GRW system provides more of a protection to process and minimise 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 system. This ensures fission and activation products generated in the core, that 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.

In addition to the above, the GRW system is pivotal to the removal of potential radioactive particulate contaminants in the air via the HVAC system to protect Plant Personnel. Particulate contaminants are filtered via High Efficiency Particulate Air (HEPA) filters before discharge.

10.2.7 Liquid Radwaste System (LRW)

The LRW is designed to protect plant personnel from radiation exposure and incorporate basic ALARP objectives and to meet the BAT requirements for radionuclide release to the environment.

The LRW collects and processes wastes produced in the plant during normal operation and anticipated operational occurrences, including shutdown, refuelling, and maintenance. The LRW consists of tanks, pumps, filters and demineralizers that processes radioactively contaminated wastes.

10.2.8 Solid Radwaste System (SRW)

The SRW collects, processes, packages, and stores radioactive solid wastes generated from normal plant operations, including Anticipated Operational Occurrences (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

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maintenance personnel ALARP. These features include remote system operation, line flushing, and shielding of components and piping containing radioactive materials.

10.2.9 Radiation Monitoring System (RMS)

The process and effluent radiological monitoring subsystem provide radiation monitoring for the relevant functions below for radiological protection.

- **Process monitors**
  - Determine concentrations of radioactive material in plant fluid systems.
  - Give early warning of a possible plant malfunction.
  - Warn operating personnel of increasing radiation/radioactivity.
  - Provide indications of radioactivity/radiation which may help to prevent inadvertent release of radioactivity to the environment.
- **Airborne monitors**
  - Airborne radiation monitors (placed in selected areas) and ventilation systems together give plant operating personnel continuous information about the airborne radioactivity levels throughout the plant. They also alert the operators of abnormally high airborne contamination concentration levels.
- **Area Radiation Monitor**
  - The area radiation monitors are in various areas of the plant, such as control room, containment building, and auxiliary building. These monitors give continuous indication and alert the operators and other station personnel to changing or abnormally high radiation conditions in the plant to prevent possible personnel overexposures and aid health physics personnel in keeping worker doses ALARP.

10.2.10 Shielding

Shielding is designed to provide plant personnel protection for all operating states, including normal full-power operation, refuelling operations, waste handling operations, in-service inspections, and plant emergency operations.

Shielding and equipment layout and design are considered in providing confidence that exposures are kept ALARP during anticipated personnel activities in areas of the plant containing radioactive materials. Systems and areas containing radioactivity or other sources of radiation are identified in the design of the generic SMR-300. Shielding is provided for these sources and locations to reduce the dose rates in line with the zoning scheme.

The design of the radiation shielding prevents propagation of radiation through shielding penetrations by:

- Minimising the area and number of straight-through paths, materials used in shielding, and the application of stepped or curved pathways or labyrinth entryways.
- Provision of shielding plugs where appropriate.
- Provision of appropriate filling material in shielding gaps, where possible.

The selection of shielding material is selected based on the nature of the radiation. Thermal stress and material degradation resulting from exposure to radiation will be considered during the selection of shielding material. In addition, where impractical to provide permanent

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shielding, distance between components and the radiation sources and temporary shields are used to reduce exposure to plant personnel.

The shielding design, including thicknesses and geometry, are determined to ensure the dose rates comply with plant radiation zoning and plant personnel exposure is ALARP.

Table 1: Safety Functions for Radiological Protection SSCs

Plant-Level Safety Function or Operational Function	Plant-level System or Group	System, Structure or Component (SSC)	Performance Requirement
Maintain Reactor Coolant Pressure and Containment Boundary Integrity Charging Isolation Degasification of the Reactor Coolant System.	Chemistry Control	Chemical and Volume Control System (Charging) (CVC)	See SDD SMR-300 System Design Description (SDD) for Chemical and Volume Control System (SV) [27]
Maintain the Reactor Coolant Pressure Boundary Integrity & 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 (PSL)	See Primary Sampling System Requirements, Open Items, and Interface Requirements Notebook [28]
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.	Main Control Room Habitability System (MCR)	Control Room Normal Ventilation System (CRNVS) Air Filtration Unit	System Design Description for Main Control Room Habitability System [29]
Containment isolation to maintain the containment boundary and vacuum relief to maintain the containment integrity.	Containment Building Ventilation (CBV) System	CBV Cooling Fan	Containment Cooling and Ventilation System [30]
Provide ventilation and temperature control to permit personnel access and to control the concentration of airborne radioactive material during normal operation.	Heating Ventilation and Air Conditioning (HVAC)	Radioactive Waste Building HVAC (RBV) System	Position Paper on HVAC Systems [31]
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 (GRW)	SDD for Gaseous Radwaste System [32]
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 (LRW)	SDD for Liquid Radwaste System [33]
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 (SRS)	SDD for Solid Radwaste System [34]

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Plant-Level Safety Function or Operational Function	Plant-level System or Group	System, Structure or Component (SSC)	Performance Requirement
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 (C&I)	Radiation Monitoring System (RMS)	Radiation Monitoring System Detector Specification [35]
Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.	**NI Civil Structures** Containment Structure (CS) Containment Enclosure Structure (CES) Reactor Auxiliary Building (RAB) Radioactive Waste Building (RWB) Intermediate Building (IB) Interim Spent Fuel Storage Installation (ISFSI)	Shielding Systems	Design Specification for Containment Structure [36] Design Specification for Containment Enclosure Structure [37] Design Specification for Reactor Auxiliary Building [38]

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10.3 CLAIMS ARGUMENTS AND EVIDENCE OVERVIEW

The primary purpose of a CAE approach is to capture the golden thread of a safety case narrative demonstrating how plant and operational evidence is brought together to justify that a high-level or fundamental claim is true. In the context of the generic SMR-300, that is how the Fundamental Purpose of the SSEC is achieved.

The Fundamental Purpose follows a golden thread throughout the SSEC to CAE via the objectives of the PSR, PER and Generic Security Report (GSR). The overarching SSEC claims are presented in Chapter A3 [2], and this chapter links to the overarching claims through Claims 2.1 and 2.2. Radiological protection is a cross-cutting topic that interfaces with many aspects of the SSEC; from the safety analysis demonstration, promulgated by Claim 2.1, to the design of systems and operational arrangements demonstrated via Claim 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 are developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

As set out in Part A Chapter 3 [2], Claim 2.1 is further decomposed across several disciplines which are responsible for assessment of the plant operation. This chapter presents the radiological protection aspects of the generic SMR-300 and therefore directly supports a claim focused on the identification of radiological protection requirements for NOs, such that they are ALARP - Claim 2.1.1.

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 2.1.1 has been further decomposed within this chapter to precisely show that the normal operating regime is controlled from the perspective of dose uptake and the demonstration of ALARP.

Claim 2.1.1.1 contributes to the Level 3 claim by identifying those radiological protection requirements that apply to NOs. This includes the identification of dose targets, constraints, and limits.

Claim 2.1.1.2 contributes to the Level 3 claim by ensuring that assessments are performed for NOs. These will be developed during the design phase, and during commissioning and operations.

Claim 2.1.1.3 contributes to the Level 3 claim by demonstrating that the assessed doses are tolerable and ALARP. This means a comparison with the relevant targets and limits, demonstrating that RGP has been followed and that further design development is not practicable to implement.

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As set out in Part A Chapter 3 [2], Claim 2.2 is further decomposed across several disciplines to consider the critical safety functions. A Level 3 claim is related to Critical Safety Function 3, Claim 2.2.3.

Claim 2.2.3: Adequate provision for the control of radiation exposure and control of release of radioactive material is incorporated into the design.

This chapter supports this Level 3 claim and is decomposed into a Level 4 claim relating to the design of the SSCs for radiological protection, Claim 2.2.3.4. This claim relates to ensuring radiation exposures are minimised by considering radiological protection in design of the SMR-300.

Claim 2.2.3.4: SSCs are designed to meet radiological protection requirements and minimise exposures.

Claim 2.2.3.4 has been further decomposed within this chapter to show that the design of SSCs important to radiological protection meets the identifies requirements and minimises exposures.

Claim 2.2.3.4.1 relates to the criticality control aspects that require designing for radiological protection to control radiation exposure.

Claim 2.2.3.4.2 relates to the design of shielding for all radioactive sources across the site, from fuel to solid, liquid, and gaseous radioactive wastes to control radiation exposure.

Claim 2.2.3.4.3 relates to the selection of materials for use in the SMR-300 and their optimisation to control and minimise radiation exposures.

Claim 2.2.3.4.4 relates to the design of ventilation and containment systems, to ensure that they control radiation exposure and the release of radioactive material.

Claim 2.2.3.4.5 relates to the design of monitoring and alarm systems, to ensure that they can detect and alarm to help to control radiation exposure and the release of radioactive material.

Claim 2.2.3.4.6 relates to the design of radiation and contamination zones, to ensure a graded approach to control of radiation exposure and of the release of radioactive material.

Table 2 shows where the decomposed claims are described in more detail in this chapter.

Table 2: CAE Decomposition

Claim No.	Claim	Subchapter
2.1.1.1	Effective dose targets, constraints and limits are identified for normal operations.	10.5 Radiological Protection Requirements
2.1.1.2	Effective doses to workers and members of the public for normal operations are assessed.	10.7 Dose Assessments for Workers and members of the Public
2.1.1.3	Effective doses to workers and members of the public during normal operations are demonstrated to be tolerable and ALARP.	10.6 Design for Radiological Protection
2.2.3.4.1	Criticality controls are implemented to ensure that criticality risks are reduced to As Low As Reasonably Practicable.	10.6.9 Criticality Controls

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Claim No.	Claim	Subchapter
2.2.3.4.2	The Generic Holtec SMR-300 shielding is designed and substantiated to minimise exposures for all plant areas and operation stages, including waste package transport.	10.6.3 Shielding Design
2.2.3.4.3	Materials selection minimises the generation of neutron activation products in SSCs	10.6.5 Material Selection
2.2.3.4.4	Ventilation and containment systems are designed to meet Radiological Protection requirements and minimise exposures.	10.6.7 Ventilation design
2.2.3.4.5	The design and layout of the Generic Holtec SMR-300 includes monitoring and alarm systems to alert to the presence of radiation and contamination.	10.6.6 Layout design
2.2.3.4.6	Radiation and contamination zoning ensures access to and between Controlled and Supervised Areas will be suitably controlled to minimise exposures to As Low As Reasonably Practicable and prevent the spread of radioactive material.	10.6.4 Area Designation, Radiation and Contamination Zoning

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10.4 CODES, STANDARDS AND METHODOLOGIES

This subchapter outlines the codes and standards used in the design of SMR-300 with respect to radiological protection.

10.4.1 Codes, Standards and Methodologies used for Radiological Protection for the SMR-300

The SMR-300 is designed in accordance with United States (US) Nuclear Regulatory Commission (NRC) regulatory guidance and applicable US Code of Federal Regulations (CFR). The standard design of the SMR-300 intends to comply with US NRC requirements set forth in U.S.NRC 10 CFR-20 Standards for Protection Against Radiation [39] and U.S.NRC Title 10, CFR Part-50 Domestic Licensing of Production and Utilization Facilities [40]

The following SMR-160 documents have been identified for familiarisation of the design and operation of a Holtec SMR, noting that that the SMR-300 is an evolution of the design, but is sufficiently similar, to allow for use of these document for information purposes:

1. HPP-160-3021, SMR-160 Design Standard for Radiation Protection (including shielding and containment) [39].
2. SMR-160 Preliminary Safety Analysis Report, Chapter 12 'Radiation Protection [40].

The International Commission on Radiological Protection (ICRP) identified the three fundamental principles of Radiological Protection in ICRP Publication 60 [41], which are maintained in ICRP Publication 103 [42]:

1. Justification of radiation exposure

No practice involving exposures to radiation should be adopted unless it produces sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes.

Any exposure to radiation must be **justified** from an economic and social standpoint and that the implementation of the generic SMR-300 will provide sufficient benefit to society to off-set the radiation detriment.

2. Optimisation

In relation to any particular source within a practice, the magnitude of individual doses, the number of people exposed, and the likelihood of incurring exposures where these are not certain to be received should all be kept as low as reasonably achievable with economic, and social factors being taken into account.

Optimisation must be applied at the design stage to ensure that design effort is devoted to ensuring collective and individual dose are minimised during NOs.

3. Limitation

The exposure of individuals resulting from the combination of all relevant practices should be subject to dose limits, or to some control of risk in the case of potential exposures.

Limitation involves the setting of worker and public radiation exposure limits for NOs.

The International Atomic Energy Agency (IAEA) Basic Safety Standards (BSS) [43] are widely adopted as the foundation for national legislation. Their purpose is to safeguard workers,

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patients, the public, and the environment from the risks associated with ionising radiation. The European Commission Directive 2013/59/EURATOM (Basic Safety Standards Directive, or BSSD) [44] is based upon the IAEA BSS and brings the international standards into European law. UK Radioactive Substances Regulations [45] and [46], Ionising Radiation Regulations 2017 (IRR17) [47],Radiation (Emergency Preparedness and Public Information) Regulations 2019 (REPPIR19) [48] and the Justification of Practices Involving Ionising Radiation Regulations 2004 [49] all stem from this directive.

The SMR-300 is designed with these three fundamental objectives to meet the US NRC Standards for Protection Against Radiation [50]. These standards are consistent with international standards for radiological protection and therefore can be considered as good practice. This chapter aims to demonstrate that the SMR-300 twin reactor unit, designed to the US regulatory scheme, can meet UK legislative requirements without any fundamental changes to the design.

The codes and standards used by Holtec for the development of the SMR-300 are summarised in Table 3. Compliance of the design with 10 CFR Part 20 [50], in consultation with the consolidated guidance to 10 CFR Part 20 (NUREG-1736) [51], is confirmed by compliance of the design and operation of the facility within the guidance of Regulatory Guides Regulatory Guides (RG) 1.8 [52], 8.8 [53] and 8.10 [54].

Table 3: SMR-300 Codes and Standards

Label	Title	Revisions
U.S.NRC Title 10, Code of Federal Regulations (10 CFR)-50	Domestic Licensing of Production and Utilization Facilities [55]	
U.S.NRC 10 CFR-20	Standards for Protection Against Radiation [50]	
US NRC 10 CFR Part 100	Reactor Site Criteria [56]	
U.S.NRC NUREG-0800	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition [57] [1]	-
U.S.NRC NUREG-1736	Consolidated Guidance: 10 CFR Part 20 — Standards for Protection Against Radiation [51]	-
U.S.NRC Regulatory Guides (RG) 1.8	Qualification and Training of Personnel for Nuclear Power Plants [52]	4 - 2021
U.S.NRC RG 1.13	Spent Fuel Storage Design Basis	2 - 2007
U.S.NRC RG 1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste [58]	3 -2021
U.S.NRC RG 1.45	Guidance on monitoring and responding to reactor coolant system leakage [59]	1 - 2008
U.S.NRC RG 1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants [60]	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 [61]	1 - 1977
U.S.NRC RG 1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants [62]	0 - 2003
U.S.NRC RG 1.196	Control Room Habitability at Light-Water Nuclear Power Reactors [63]	1 - 2007
U.S.NRC RG 1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors [64]	0 - 2003

[1] Chapter 12 of NUREG 0800 – Radiation Protection [57] contains sections on Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable, Radiation Sources, Radiation Protection Design Features, and Operational Radiation Protection Program. These documents are broadly equivalent to the Office for Nuclear Regulation Technical Assessment Guides and Technical Inspection Guides.

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Label	Title	Revisions
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 [53]	3 - 1978
U.S.NRC RG 8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable [54]	2 - 2016

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

UK legislations for nuclear facilities and radiological protection stem from international standards, as discussed in 10.4.1.

Further development work planned during the GDA will highlight the similarities and gaps between US regulations and guidance and the UK regulatory requirements, codes of practice and guidance. The US-UK Regulatory Framework Document will also summarise methods for addressing gaps in the design to inform ALARP/BAT optioneering studies and development of the generic SMR-300, to ensure doses and risks to workers and MoP are ALARP.

The primary UK legislation for radiological protection is IRR17 [47] and the associated Approved Code of Practice (ACOP) and guidance [65]. These set out the minimum legal duties for work with ionising radiation, providing RGP, which, if followed, should be sufficient to demonstrate that exposure to ionising radiation arising from work activities is reduced to ALARP.

The Nuclear Installations Act 1965 (NIA) [66] sets out the licensing regime, which includes the attachment of conditions to nuclear site licences. The licence covers design, construction, commissioning, operation, and end of life/decommissioning for a nuclear power facility. Licence Condition 18 is concerned with Radiological Protection; specifically, the requirement to assess average effective doses to defined classes of persons and to notify the Office for Nuclear Regulation (ONR) when these average effective dose equivalent values have exceeded a level the ONR may specify.

Additional regulations of relevance to this chapter also include:

- The Health and Safety at Work etc. Act 1974 [67], defines the structure and authority for the regulation and enforcement of workplace health, safety, and welfare. It outlines the responsibilities of both the employer and employee to ensure risks are reduced SFAIRP to employees and others not in their employ, ensuring there is a safe working environment.
- The Management of Health and Safety at Work Regulations 1999 [68] forms the basis for the requirement to perform, record and maintain risk assessments. The requirement for radiological risk assessment in IRR17 stems from this legislation.
- The Justification of Practices Involving Ionising Radiation Regulations 2004 [49] require any new use of ionising radiation to be justified, and that the benefits of the use must outweigh the risks. The regulations also require that any new use of ionising radiation must be authorised by the relevant Justifying Authority, e.g., the Secretary of State.
- The REPPIR 2019 [48] and associated ACOP and guidance [69] set out the requirements for planning for and managing the consequences of radiation emergencies arising from work with ionising radiation.

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- The Environmental Permitting (England and Wales) Regulations 2016 (EPR16) [45] are concerned with protecting the public and the environment from radioactive substances. Within EPR16 there is a requirement to assess exposures to MoP from direct radiation from onsite sources. This will be addressed in this Chapter as well as in support of the RIA in PER Chapter 3 [4].

The key Safety Assessment Principles (SAPs) relating to radiological protection listed in the ONR SAPs for Nuclear Facilities [70] are:

- RP1 (Normal operation (Planned Exposure Situations)) - Adequate protection against exposure to radiation and radioactive substances should be provided in those parts of the facility to which access is permitted during NO.
- RP2 (Fault and accident conditions (Emergency Exposure Situations)) - Adequate protection against exposure to radiation and radioactive contamination should be provided in those parts of the facility that will need to be accessed during faults or as part of accident management. This should include prevention or mitigation of accident consequences.
- RP3 (Designated areas) - Where appropriate, designated areas should be further divided, with associated controls, to restrict exposure and prevent the spread of radioactive material.
- RP4 (Contaminated areas) - Effective means for protecting persons entering and working in contaminated areas should be provided.
- RP5 (Decontamination) - Suitable and sufficient arrangements for decontaminating people, the facility, its plant, and equipment should be provided.
- RP6 (Shielding) - Where shielding has been identified as a means of restricting dose, it should be effective under all NO and fault conditions where it provides this safety function.
- RP7 (Hierarchy of control measures) - The duty holder should establish a hierarchy of control measures to optimise protection in accordance with IRR17.

Criticality safety principles apply to the processing, handling or storage of fissile materials in significant quantities with respect to the minimum critical mass, and in locations where criticality is not intended (i.e., outside of the reactor core). The key SAPs relating to criticality safety listed in the ONR SAPs for Nuclear Facilities [70] are:

- ECR.1 (Engineering principle: criticality safety (Safety measures)) - Wherever a significant amount of fissile material may be present, there should be safety measures to protect against unplanned criticality.
- ECR.2 (Engineering principle: criticality safety (Double contingency approach)) - Criticality safety cases should employ the double contingency approach.

Chapter B10 Radiological Protection should be read in conjunction with the fault analysis chapters, B14 [6], B15 [17], B16 [18], B21 [15] and B22 [16] to identify any potential for faults that can challenge criticality safety.

In addition to these radiological protection and criticality principles, NTs are provided for NOs of nuclear facilities:

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- Numerical Target NT.1 (Assessment against targets) - Safety cases should be assessed against the SAP's NT for normal operational, design basis fault and radiological accident risks to people on and off the site.

This chapter is concerned with Targets 1, 2 and 3, which are identified in PSR Part A Chapter 2 'General Design and Site Characteristics' [8] and stated again in subchapter 10.5.1.

The following ONR Technical Assessment Guides (TAG) and Technical Inspection Guides (TIG) are also applicable:

- NS-TAST-GD-038 – Radiological Protection [71].
- NS-TAST-GD-002 – Radiation Shielding [72].
- NS-TAST-GD-004 – Fundamental Principles [73].
- NS-TAST-GD-005 – Guidance on the Demonstration of ALARP [74].
- NS-TAST-GD-022 – Ventilation [75].
- NS-TAST-GD-024 – Management of Radioactive Material and Radioactive Waste on Nuclear Licensed Sites [76].
- NS-TAST-GD-027 – Training and assuring personnel competence [77].
- NS-TAST-GD-041 – Criticality Safety [78].
- NS-TAST-GD-043 – Radiological Analysis – Normal Operation [79].
- NS-TAST-GD-045 – Radiological Analysis – Fault Conditions [80].
- NS-TAST-GD-051 – Purpose Scope and Content of a Safety Case [81].
- NS-TAST-GD-100 – Shielding and Dose Rate Safety Assessment of Transport Packages [82].
- NS-INSP-GD-054 – The Ionising Radiations Regulations 2017 [83].

10.4.2.1 Key Definitions and Regulations within IRR17 for Reactor Design

The ACOP [65] to IRR17 [47] holds a special legal status in the UK. Through following the ACOP, and associated guidance, a radiation employer has evidence to demonstrate that doses will be reduced to ALARP.

To demonstrate compliance with UK legislation (as identified in the regulatory framework document), design aspects of the SMR-300 will be mapped against the regulatory requirements in the IRR17, SAPs and TAGs Compliance Matrix. This document will also present how the ACOP will be addressed, including: area designation and radiation/contamination zoning; dose management, limits and constraints; classification and monitoring of workers; discussion regarding training and Suitably Qualified and Experienced Persons (SQEP); radiation risk assessment; and restriction of exposure.

Throughout this document, workers are denoted as On-Site Workers (OSWs). At this design stage OSWs is used as a blanket term to cover all persons employed to work on the site, whether or not they work with radiation, are designated as classified workers (See Regulation 21 below), are directly employed or outside workers.

The following specific regulations are especially relevant at the design stage for a new facility.

- Regulation 9 Restriction of exposure

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Radiological risk will be considered, through review of other PSR chapters regarding faults/scenarios (i.e., PSR Chapters B14 'Safety and Design Basis Accident Analysis' [6], B15 'Beyond Design Basis, Severe Accident Analysis, and Emergency Preparedness' [17], B16 'Probabilistic Safety Analysis' [18], B21 'External Hazards' [15] and B22 'Internal Hazards' [16]) to support the establishment of the use of hierarchy of controls. This will require a review of where access controls, shielding, other engineered controls, design features and safety features are claimed to limit exposures to OSWs and MoP. It is anticipated that this will be performed at detailed design stage when operator occupancy is better understood.

- Regulation 17 Designation of controlled or supervised areas

See separate discussion of this in sub-chapter 10.6.4.

- Regulation 20 Monitoring of designated areas

Monitoring will be covered in the IRR17, SAPs and TAGs Compliance Matrix and summarised in sub-chapter 10.6.8.

- Regulations 21 Designation of classified persons and 22 Dose assessment and recording

Classification of persons and dosimetry arrangements will be covered in the IRR17, SAPs and TAGs Compliance Matrix and summarised in this sub-chapter.

10.4.3 Lessons Learnt

Operating Experience (OPEX) on radiation doses to workers from comparable operations at other Pressurised Water Reactor (PWR) facilities can be found on the following websites:

- European Platform for Occupational Radiation Exposure [84]
- World Association of Nuclear Operators – Performance Indicators [85]
- Information System on Occupational Exposure – Annual Reports [86]
- Institute for Radiation Protection and Nuclear Safety – Reports of Expertise [87]
- REIRS: Annual Report on Occupational Radiation Exposure, NUREG-0713 [88]

The US NRC Human Factors Engineering Program Review Model, NUREG-0711, Revision 3 [89] includes a section on Operational Experience Review (OER). Compliance with NUREG, including a strategy for addressing each of the sections in NUREG 0711 [89] from the early stages of design, is part of the US NRC licensing process. The SMR-300 is being designed in line with this process to allow it to be licensed in the US. Task analysis is part of the HF Engineering process and is used to inform design decisions. The level of detail available for GDA will be dependent on safety significance and the maturity of the design. The HF Engineering team, carry out a review of incidents from US power stations and World Association of Nuclear Operators (WANO)/Institute of Nuclear Power Operations (INPO) which they use to inform task analysis and functional requirements analysis and ensure good practice and lessons learnt from similar facilities are incorporated are incorporated into the design, . The OER relates to any incidents/learning from experience, not just HF. More on this can be found within PSR Chapter B17 [90].

IAEA Specific Safety Guide-90 Radiation Protection Aspects for Nuclear Power Plants [91] has recently been published. The guide aims to provide recommendations for ensuring

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radiation protection in the design of new nuclear power plants, in design modifications to operating plants and in checking of the adequacy of the design at different stages in the lifetime of operating plants. The guide emphasises the importance of including operating experience in the development of the design to minimise exposures.

A review of relevant Regulatory Issues (RIs) and Regulatory Observations (ROs) raised for previous GDA submissions has been completed. All ROs raised under Radiological Protection are presented in Appendix A, and RIs in Appendix B. This review allowed learning from the experiences in earlier GDA submissions, to give a greater understanding of the regulatory focus within radiological protection, and to seek to develop these topic areas at the earliest opportunity.

- Relevant ROs and RIs relating to source terms.
    - RI-ABWR-0001 2nd June 2015 - Definition and justification for the radioactive source terms in UK Advanced Boiling Water Reactor (ABWR) during Normal Operations [92].
    - RO-ABWR-0006 1ST July 2015 Source Terms [92].
- Previous GDA ROs and RIs relevant to ALARP.
    - RO-ABWR-0073 16th August 2016 [92] - Robust demonstration that the design of the UK ABWR off-gas system reduces risks SFAIRP.
- Previous GDA ROs and RIs relevant to source term minimisation.
    - RO-UKHPR1000-0026 9th Oct 2020 [92] - Demonstration that radioactivity has been reduced to SFAIRP.
    - RO-UKHPR1000-0049 14th August 2020 [92] - Generation, transport and behaviour of tritium during normal operations.
    - RO-UKHPR1000-0050 29th September 2020 [92] - Selected spent fuel interim storage technology ALARP demonstration.
- Previous GDA ROs and RIs relevant to shielding design.
    - RO-UKHPR1000-0028 16th Dec 2019 [92] – Adequate justification of estimated public doses for UK Hualong one Pressurised Reactor (HPR)1000.
    - RO-ABWR-0065 20th Oct 2015 [92] – Demonstration of adequate design and implementation of inherently safe techniques and structures to minimise radiation dose rates via through wall penetrations during all operating modes and for the lifetime of the facility, whilst being cognisant of design requirements relating to other discipline areas.
    - RO-UKHPR1000-0060 2nd Mar 2021 [92] – Scope and plan for radiation shielding assessments for the UK HPR1000 Generic Design.
- Previous GDA ROs and RIs relevant to radiation and contamination zoning.
    - RO-ABWR-0064 23rd October 2015 [92] - Design approach to identification and provision of both permanent and temporary features necessary for the adequate control of radioactive contamination across the full lifetime of UKABWR.
- Previous GDA ROs and RIs relevant to material selection.
    - RO-ABWR-0035 3rd October 2014 [92] - Robust justification for the materials selected for UK ABWR.
- Previous GDA ROs and RIs relevant to chemistry control.
    - RO-ABWR-0072 15th August 2016 [92] Suitable and sufficient consideration of chemistry control during UK ABWR commissioning.
- Previous GDA ROs and RIs relevant to system and equipment design.

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  - RO-ABWR-0054 15th May 2015 [92] UK ABWR – Chemical/process engineering design approach.
  - RO-ABWR-0022 17th October 2014 [92] UK ABWR - Demonstration that the primary cooling system operating chemistry reduces risks SFAIRP.
- Previous GDA ROs and RIs relevant to Ventilation.
  - RO-ABWR-0017 16th September 2014 [92] - Nuclear ventilation codes and standards.
  - RO-ABWR-0075 22nd November 2016 [92] - Robust demonstration that the design of the UK ABWR HVAC system has been adequately conceived and reduces risks SFAIRP.
- Previous GDA ROs and RIs Relevant to Decommissioning.
  - RO-UKHPR1000-0042 29th April 2020 [92] - Robust demonstration of ALARP for decommissioning of the UK radiological protection claims, arguments, evidence.

Lessons learnt from these ROs and RIs are discussed in Appendix A and Appendix B respectively.

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10.5 RADIOLOGICAL PROTECTION REQUIREMENTS

Claim 2.1.1.1: Effective dose targets, constraints and limits are identified for normal operations.

The SMR-300 design approach is to minimise radiation exposure to plant personnel and the public with due consideration for HF. The ALARA principle is applied throughout the design of the SMR-300 to minimise radiation doses and the release of radioactive materials into the environment. This chapter aims to demonstrate that designing the SMR-300 twin reactor unit to the US regulatory scheme will allow the facility to achieve UK legislative requirements.

The radiological protection philosophy and measures will be developed as part of the generic SMR-300 design, demonstrating that these measures ensure NO doses are ALARP. The key objectives are to:

- Develop the structure of the radiological protection claims.
- Derive arguments that substantiate claims or identify references to supporting documentation.
- Present radioactive source terms that have been considered.
- Develop the strategy to ensure exposure is ALARP.
- Define the SSCs for radiological protection control measures and review their adequacy.
- Describe radiation and contamination monitoring systems for occupational exposure and review their adequacy.
- Describe the dose assessment methodology for MoP from direct radiation.
- Describe the internal and external dose assessment methodology for OSWs.
- Describe the control measures used to prevent criticality from new and used fuel during handling and storage.
- Identify links to relevant content of GDA Regulatory Observations (RO) and Issues (RI) raised against previous GDA.

This chapter will describe the design approaches to radiological protection for all phases of the facility lifetime: design, construction, commissioning, all modes of operation and decommissioning; Concerning exposures to individuals, teams, and MoP during routine work and consideration of ALARP demonstration requirements with regards to:

- Risk of exposure of OSWs and MoP to Ionising Radiation (IR) from NOs allowing for inclusion of all planned outage types, inclusive of relevant EIMT activities.
- Ensuring regulatory requirements are met through the minimisation of radiation and contamination levels, achieved by applying Radiological Protection principles to equipment systems and layout.
- To provide confidence that the design will not result in radiological doses greater than Basic Safety Level (BSL) targets, and OPEX from comparable reactor operations.
- Demonstrate relevant controls are in place to prevent criticality from new and used fuel during handling and storage.

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10.5.1 Dose Limits and Targets

The dose acceptance criteria for the standard US SMR-300 plant design comply with US dose limitation requirements, as explained in PSR PART A Chapter 2 'General Design and Site Characteristics' [8]. These are reproduced in Table 4 and Table 5 for clarity.

Table 4: Effective Dose Limits for Normal Operation (US)

Class of Person	Period	Effective Dose in mSv (rem)
Nuclear energy worker	a) One-year dosimetry period	50 (5)
	b) Five-year dosimetry period	100 (10)
Declared pregnant nuclear energy worker	Balance of the pregnancy after declaration	4 (0.4)
A person who is not a radiation worker (member of public)	One calendar year	1 (0.1)

Table 5: Equivalent Dose Limits for Normal Operation (US)

Organ or Tissue	Person	Period	Equivalent Dose in mSv (rem)
Lens of an eye	Nuclear energy worker	One-year dosimetry period	150 (15)
Skin (averaged over 1 cm2 that receives highest equivalent dose)	Nuclear energy worker	One-year dosimetry period	500 (50)
Hands and feet	Nuclear energy worker	One-year dosimetry period	500 (50)

10.5.1.1 SMR-300 Design Dose Constraints

Application of the ALARA principle in the US begins with the establishment of design dose targets based on OPEX and best nuclear industry practices.

Annual collective Total Effective Dose Equivalent (TEDE) per reactor for commercial PWR is shown to be averaging below 0.5 person-Sv/y (50 man-rem/year) [88] and are trending downwards.

To reduce doses to OSWs and MoP to levels that are ALARA and pursue the best practice, design dose targets (constraints in IRR17 parlance) for NOs have been established. These were derived for the SMR-160 in [39] and will be replicated in the upcoming SMR-300 Radiation Protection Design Standard.

a. The station collective occupational dose is less than or equal to 0.25 person-Sv/y (25 man-rem/year) averaged over the reactor life cycle of a single unit.
b. The effective occupational radiation dose of plant personnel designated as Radiation Worker is less than or equal to 10 mSv (1 rem) per year unless with

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written consent[2], can be increased to less than or equal to 20 mSv (2 rem) per year.

c. For members of the public, the average target dose to the critical groups of the population at or beyond the exclusion area boundary is less than or equal to 10 µSv/y (1 mrem/year), i.e., 1% of the regulatory limit over the lifetime of the Plant.

These design dose targets are considered an upper level of dose and the design philosophy is to better them wherever possible. The US design targets are comparable to other modern reactor designs and are below UK dose limits. Design of the SMR-300 will ensure that doses to OSWs and MoP are ALARA. ALARP will be applied when assessing the design against UK regulatory expectations.

10.5.1.2 Generic SMR-300 UK Dose Limits and Targets

US and UK legislation differ in worker radiation exposure limits, which informs the worker arrangements. The limits and targets presented in Table 4 and Table 5 are different from those expected in the UK, as specified in IRR17 [47].

IRR17 Regulation 12 [47] – Dose Limitation States:

> *Every employer must ensure that its employees and other persons within a class specified in Schedule 3 are not exposed to ionising radiation to an extent that any dose limit specified in Part I of that Schedule for such class of person is exceeded in any calendar year.*

These dose limits are summarised in Table 6.

Table 6: Dose limits for classes of person

Class of Person	Effective or equivalent dose type	Effective Dose Limit (Calendar Year unless stipulated)
Employee or Trainee	Whole Body	20 mSv 100 mSv over 5 years, max 50 mSv in any single calendar year[3]
	Lens of the Eye	20 mSv 100 mSv over 5 years, max 50 mSv in any single calendar year[3]
	Skin and Extremities	500 mSv
Trainees aged under 18 years	Whole Body	6 mSv
	Lens of the Eye	15 mSv

[2]A target can be exceeded if the excess can be justified, the upper boundary target is equivalent to the UK legal limit for radiation workers, and is therefore the upper limit for individual dose uptake in the UK. Doses should be reduced to below the target to ensure doses are (US) ALARA.

[3] In accordance with conditions approved by the Health and Safety Executive

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Class of Person	Effective or equivalent dose type	Effective Dose Limit (Calendar Year unless stipulated)
	Skin and Extremities	150 mSv
Other Persons	Whole Body	6 mSv
	Lens of the Eye	15 mSv
	Skin and Extremities	50 mSv

Numerical dose targets are presented in PSR PART A Chapter 2 'General Design and Site Characteristics' [8]. The basis for this is the ONR SAPs [70], where NT Principle 1 (Assessment against targets) states that safety cases should be assessed against the SAPs NT for normal operational, design basis fault and radiological accident risks to people on and off the site. The targets of interest to NOs are Targets 1-3.

- Target 1 provides the BSL which is also the Legal Limit (LL) and the Basic Safety Objective (BSO) for any person on the site.
- Target 2 details the BSL and BSO for any group on the site.
- Target 3 details the BSO and BSL for MoP and other persons off the site.

The targets for NOs radiological dose are also documented in NS-TAST-GD-043 – Radiological Analysis – Normal Operation [79] and replicated in Table 7.

Table 7: Normal Operations Dose Targets

Target	Classification	Dose Target	Effective Dose Limit (Calendar Year)
1- Any person on the site	Employees working with Ionising Radiation	BSL (LL)	20 mSv
		BSO	1 mSv
	Other employees on the site	BSL	2 mSv
		BSO	0.1 mSv
2-Any group on site	Average effective dose to defined groups of employees	BSL	10 mSv
		BSO	0.5 mSv
3- Any person off the site	Any person off the site	BSL	1 mSv
		BSO	0.02 mSv

10.5.2 ALARP in Radiological Protection and Dose Limitation

Given the discrepancy in exposure limits between US and UK, meeting the requirements of UK legislation may require some changes to the design. These changes must be evaluated. This sub-chapter describes the strategy to ensure that external and internal exposures to

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OSWs and MoP during NOs are within legal limits and reduced to ALARP through design development of the generic SMR-300 and comparison against design dose constraints (targets in US parlance).

As the generic SMR-300 is developed, the principle of ALARP should be embedded into design decisions and processes; ensuring that records are documented as arguments and evidence in support of decisions reducing risk to ALARP. This should continue through change control, as the design matures.

The information recorded as part of the optioneering process should include a comprehensive review of RGP and OPEX to support identification of options, and evaluation of options in consideration of the radiological protection impacts.

Key to the ALARP demonstration of ALARP in radiological protection is application of the Hierarchy of Control (HoC), as defined in the IRRs 2017 [47] and documented below, which is implemented when evaluating the benefits and detriments of each design option, to ensure passive engineered safety features are prioritised rather than active systems or operator action.

In meeting the key requirements of ALARP in radiological protection, it is fundamental that design is optimised to ensure both external and internal exposures to radiation dose uptake by OSWs and MoP are minimised. Although the key focus on this Chapter is ALARP from NOs, full optimisation of design is necessary to take account of the requirements of BAT to ensure radiological discharges and radioactive waste are minimised along with the requirements to meet ALARP. In essence the requirements of BAT and ALARP must work in synergy, as discussed in Holtec SMR GDA PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [19] and detailed in HPP-3295-0017, Holtec SMR-300 Generic Design Assessment Reference Design Process and GDA Prospective Design Change Register [93].

10.5.2.1 ALARP in Principle

The fundamental principles of ALARP in radiological protection in the UK is documented in IRR17 Regulation 9 – Restriction of Exposure [47] which states in paragraphs (1) and (2):

> *9 (1) Every employer must, in relation to any work with ionising radiation that it undertakes, take all necessary steps to restrict SFAIRP the extent to which its employees and other persons are exposed to ionising radiation.*
>
> *9 (2) Without prejudice to the generality of paragraph (1), an employer in relation to any work with ionising radiation that it undertakes must –*
>
> > *(a) so far as is reasonably practicable achieve the restriction of exposure to ionising radiation required under paragraph (1) by means of engineering controls, design features and by the provision and use of safety features and warning devices;*
> >
> > *(b) provide such systems of work as will, so far as is reasonably practicable, restrict the exposure to ionising radiation of employees and other persons; and*

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> *(c) where it is reasonably practicable to further restrict exposure to ionising radiation by means of personal protective equipment, provide employees or other persons with adequate and suitable personal protective equipment (including respiratory protective equipment) unless the use of personal protective equipment of a particular kind is not appropriate having regard to the nature of the work or the circumstances of the particular case*

Thereby introducing in law, the requirement for the Hierarchy of Control Measures (HoCMs).

SAP RP.7 (Hierarchy of control measures) [70] states:

> *The dutyholder should establish a HoCMs to optimise protection in accordance with IRR17.*

The requirements of Regulation 9 (2) and RP7, the HoCMs, are presented in Table 8 Review of the design will be carried out to demonstrate that the HoCMs are inherent in the design, thereby deemed to be applied to restrict exposure to ionising radiation SFAIRP. The prescriptive nature of US legislation requires process and/or engineering controls to be considered before other controls and PPE. For example, Sub-part H of 10 CFR 20 [50] requires the use of process or engineering controls to control the concentrations of radioactive material in air (20.1701), where these are not practical, other measures should be applied (20.1702). Examples of how this is implemented in the design will be presented.

Table 8: Hierachy of Control Measures

Control Measure	Hierarchy	Examples
Engineering Controls		Design features, safety features and warning devices
Systems of Work		Procedural controls, Written Systems of Work, Local Rules etc
PPE including RPE		Coveralls, hats, caps, gloves, respirators, airhoods, overshoes

The detailed hierarchy of hazard controls (beyond RP7) is discussed in 10.6.1.

In summary, the exposure of OSWs to radiation in the SMR-300 design is limited and maintained ALARP by:

- Minimising (and eliminating where possible) the production and build-up of radionuclides. See also subchapters 10.6.1 and 10.6.5.
- Employing advanced design principles for local plant features including plant layout, RM containment, shielding design, and the design of systems and components for reliability and maintainability.
- Controlling access to areas of high activity or radiation, or where potential for possible contamination exists.

In addition, the design provides for auxiliary features (i.e., for storage and donning of personal protective clothing) and decontamination facilities where required.

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Control of radiation exposures to OSWs and MoP is achieved by a combination of measures incorporated into plant design, and by adherence to a set of approved procedures and limits when the plant is in operation.

The ONR TAGs on Radiological Protection [71] and ALARP [74] provide guidance on the implementation of ALARP; additionally, representatives from UK nuclear operators (Industry Radiological Protection Co-ordination Group, on behalf of the Safety Directors Forum) have produced a Good Practice Guide [94]. Implementation of these RGP should be adopted in progression of the generic SMR-300 design to demonstrate that radiological risk and exposures are ALARP.

10.5.3 CAE Summary

The dose limits and targets for the SMR-300 design have been developed to comply with the US ALARA principle. Compliance with the ALARA principle in the US is achieved by establishing design dose targets that are lower than the US limits, based on OPEX and best nuclear industry practices. This degree of conservatism allows the SMR-300 to meet the US defined ALARA principle and reduces worker dose as priority.

US and UK legislation differ in worker radiation exposure limits, which informs the worker arrangements. The application of the ALARP principle and the IRRs are the main changes that will drive any design decisions for the generic SMR-300. Radiological protection requirements have been identified in accordance with RGP in the UK. This includes design constraints, dose limits and targets and design principles, such as the HoCMs, thereby meeting Claim 2.1.1.1.

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10.6 DESIGN FOR RADIOLOGICAL PROTECTION

Claim 2.1.1.3: Effective doses to workers and members of the public during normal operations are demonstrated to be tolerable and ALARP.

Claim 2.2.3.4.1: Criticality controls are implemented to ensure that criticality risks are reduced to As Low As Reasonably Practicable.

Claim 2.2.3.4.2: The Generic Holtec SMR-300 shielding is designed and substantiated to minimise exposures for all plant areas and operation stages, including waste package transport.

Claim 2.2.3.4.3: Materials selection minimises the generation of neutron activation products in SSCs.

Claim 2.2.3.4.4: Ventilation and containment systems are designed to meet Radiological Protection requirements and minimise exposures.

Claim 2.2.3.4.5: 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 2.2.3.4.6: Radiation and contamination zoning ensures access to and between Controlled and Supervised Areas will be suitably controlled to minimise exposures to As Low As Reasonably Practicable and prevent the spread of radioactive material.

Considering the radiological protection requirements identified above, this subchapter sets out further design provisions to demonstrate that risks are reduced to ALARP.

The safety categorisation and classification methodology currently defined for SMR-300 by Holtec International utilises the U.S. NRC regulatory guide 1.26, Revision 6 [95] and related guidance documents. Radiological protection SSCs have been classified according to this process and their safety functions determined. The methodology for safety function categorisation and classification to meet UK requirements is discussed in Chapter A2 [96].

10.6.1 ALARP in Design Practice

The minimisation of radiation exposures and plant activity levels is achieved through optimisation of plant radioactive inventory; shielding provisions; layout and system design; criticality controls, and monitoring. In general, the following design features are applicable for ensuring doses are ALARP:

1. Source term minimisation – see subchapter 10.6.2.
2. Shielding design - see subchapter 10.6.3.
3. Area designation including radiation and contamination zoning - see subchapter 10.6.4.
4. Material selection (to limit activation products) - see subchapter 10.6.5, including;
    a. Chemistry control.
    b. System and equipment design.
5. Layout design - see subchapter 10.6.6.

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6. Ventilation design - see subchapter 10.6.7.
7. Radiation and contamination monitoring - see subchapter 10.6.8.
8. Criticality controls - see subchapter 10.6.9.
9. Review of radiation protection claimed in accident protection – see subchapter 10.6.10.

The predominant way to ensure radiation doses to OSWs and MoP meet BSLs, BSOs and design dose constraints, and are maintained ALARP, is to reduce the hazard SFAIRP in the first instance using the Eliminate, Reduce, Isolate, Control, Personal Protective Equipment, Discipline (ERIC/PD) methodology. This is presented in Figure 1.

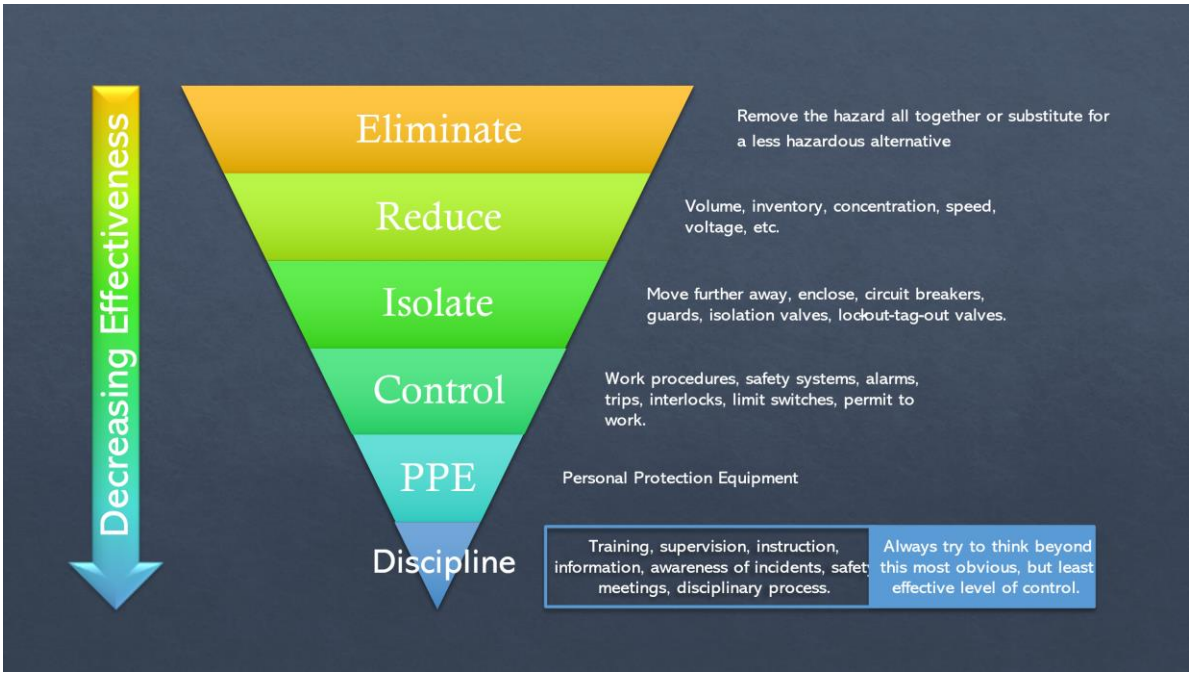

Figure 1: ERIC/PD infographic

10.6.2 Source Term Minimisation

There are two main types of source term to be considered – NOs and accident source terms. Accident source term minimisation is excluded from the scope of this chapter.

Holtec SMR-300 GDA PSR PART B Chapter 23 ‘Reactor Chemistry’ [24] describes the source term from the reactor primary and secondary coolant, material selection and optimisation to reduce doses SFAIRP.

The NOs source term is generated from a series of conservative assumptions about the operation of the reactor and the contained sources of radiation. Normal operations design basis source terms have been considered for:

1. The core source term -reactor fuel assemblies neutron and gamma source term including the neutron and gamma flux, decay heat and radionuclides.
2. Primary coolant source term including fission products, activated corrosion products and coolant activation products activity concentrations.
3. Secondary coolant source term produced from a leakage estimation from the primary coolant.

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4. RCS components source terms, including hot sections, cold sections, average temperature sections and the pressuriser.
5. CVC System components source terms, including the mixed and cation bed demineralisers, reactor coolant filters and regenerative and non-regenerative heat exchangers.
6. LRW system source terms, including tanks, pumps, filters, demineralisers and a degasifier that store or process liquid radwaste from the equipment drain, floor drain and chemical waste subsystems.
7. SRW system source terms, including spent resin tanks, resin pumps that store or process wet solid wastes, dry solid wastes and miscellaneous solid wastes.

For the SMR-160, the relevant documents were 'SMR-160 Core Isotopics and Source Terms' [97] and 'Contained Radiation Source for Normal Operations' [98]. These documents are due to be updated and presented as the design reference in PSR Chapter A2 [8].

The NOs source term will be defined for the SMR-300 and used as an input to:

- Radiation shielding design calculations.
- Radwaste systems design.
- Effluent concentrations.
- Occupational dose estimations.
- Radiation protection assessment, including personnel protection.
- Ventilation systems design.

The conservative assessment of the source terms for NOs and accident conditions highlights areas where there is room for improvement. The following highlights the ways in which the source term is minimised in the design:

1. To minimise deposits in the RCS and those systems that connect to the RCS by the following methods:
    a. Materials selection: Selection of materials to minimise the generation of key activation products, including ^{58}Co and ^{60}Co corrosion products, which constitute substantial radiation sources in the deposits, see subchapter 0.
    b. Chemistry control: Minimising the creation of the corrosion and erosion products (deposits) by selection of adequate materials and control of the water chemistry to minimise the activation of coolant, see subchapter 10.6.5.1.
    c. System and equipment design: Many design features, for example, cleaning systems for effective minimisation of impurities and corrosion products that could form deposits and minimising deposit traps, minimising air in the coolant by degassing to minimise ^{41}Ar production, see subchapter 10.6.5.2.
2. Improving spent fuel management and storage by the following methods:
    a. Chemistry control: Water in the SFP shall be maintained at a low activity level by means of a purification system with particulate filters and ion exchangers. This also covers filtration for damaged or failed fuel stored in the SFP.
    b. The integrated fuel management strategy: Utilises on-site dry storage once sufficient cooling of spent fuel has occurred, limits the accumulation of spent fuel within the SFP and the liquid source term in the SFP and down-stream resins and filters.
    c. System and equipment design: Fuel defects reduced SFAIRP to minimise potential of alpha contamination hazard from tramp uranium. Tramp uranium

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is the term used to define uranium and fissile contamination of the in-core portion of the primary heat-transport system. The tramp uranium comes from uranium contamination on the surfaces of fuel rods and from fuel defects and could be found in the fuel assemblies, SFP, and steam generator room. Damaged-fuel containers for use within the SFP incorporate filtration systems for any such instances of fuel damage to prevent any radioactive material from leaking into the pool water.

3. The selection of construction materials and the method of construction will aim to reduce the generation of activation products within these structures, through for example, the use of low-europium concretes, low-cobalt steels, and positioning of rebar within concrete structures.

As radiological protection is concerned with all sources of radioactivity on the site (and discharged from the site), an overarching Source Term Report will be produced, which will be referenced from all other disciplines (including radioactive waste, fuel, decommissioning, quantification of discharges, reactor chemistry, etc.). Design basis and best estimate source terms will be presented as required for all topic areas.

10.6.3 Shielding Design

Shielding is designed to provide protection to OSWs (and MoP) for all operating and shutdown states, including normal full-power operation, refuelling operations, waste handling operations, in service inspections, accident conditions and decommissioning.

Computer codes based on point kernel and Monte Carlo methods are used to calculate gamma and neutron dose rates from onsite radiation sources. Details of the methods, considerations and constraints assumed for the shielding analysis will be included in a Shielding Basis of Design document. This document will describe the technical approach to shielding design planned for the project. It will define the dose constraints that shielding assessment will be working against (making reference to occupancy and radiation zoning), assumptions and definitions (such as the sources to be assessed and materials specifications), modelling methods (including any codes to be used) and the use of the Source Term Report. The shielding design for the SMR-300 will then be optimised through utilisation of this data in combination with radiation zoning schemes and predicted occupancy modelling.

Within the shielding design for SMR-300 the following will be implemented or taken into account[4]:

- The neutron and gamma flux from the reactor core is reduced by shielding provided by reactor internal components, notably the neutron reflector, and reactor vessel.
- Process piping for reactor coolant outside of the core is arranged and shielded appropriately, minimising routing through personnel access corridors.
- The shielding design of the CVC System components will be based on the system containing reactor coolant to a design basis source term.

[4] This is a list of shielding considerations for the SMR-160 [39] [40] design. A similar review for the SMR-300 is underway.

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- Refuelling Pool and Refuelling Cavity are designed for a flood-up approach based on RGP and OPEX, to provide bulk water shielding and minimise dose rates during refuelling activities, with upper internals, fuel, and core barrel removed under water.
- A shielding arrangement may be implemented to protect against the radiation hazard from build-up of deposits in the SFP if mitigations to minimise the build-up are insufficient to reduce doses to ALARP.
- Shielding for each component of the liquid radioactive waste system is based on radioactivity found in each of the system components; for example, this will include shielding components for high activity resins, and radiologically contaminated sludge expected in storage tanks.
- For solid radioactive waste, the shielding and layout of such areas are considered during design of the facility. Higher activity sources, such as spent resin, are processed in a separate program.
- Minimisation of the surface area of penetrations, and number of straight-through paths, materials used in the shielding of penetrations, and consideration of utilising stepped or curved pathways or labyrinth entry ways.
- Provision of shield plugs where necessary; and provision of appropriate filling material in shielding gaps, where possible.
- Shielding design for spent fuel transfer equipment, such as the HI-TRAC overpack transfer cask, and for the UMAX system as described in PSR Chapter B11 [99] reduces doses to OSW and MoP. Shielding is provided through the use of steel, lead and water. The additional shielding afforded by the below grade HI-STORM UMAX system configuration reduces doses to both OSWs and MoP to extremely low levels.

In developing the Generic SMR-300, for deployment in the UK, consideration of these measures will be included as part of the ALARP assessment, to determine whether dose reduction through the provision of further shielding is reasonably practicable. Additionally, temporary shielding will be reviewed to determine whether its use represents ALARP, taking into consideration dose and risk associated with the installation and removal of the shielding.

The SMR-300 plant is shielded to reduce the contribution of direct radiation dose. Design aim for the facility is to have no detectable radiation dose (above background) at the site boundary such that shielding of the SMR-300 is sufficient that direct radiation dose at the site boundary is insignificant to MoP (the SMR-300 design target for MoP is 0.01 mSv/y) and therefore the Generic SMR-300 should meet Target 3 BSO [70] and ensure exposures are ALARP. Radiation shielding for MoP from direct radiation is primarily provided by outer walls and roofs of buildings. This is further supported by the design incorporating a below grade arrangement, where much of the reactor building (including the reactor and spent fuel pool) is positioned below ground level. Additionally, the design incorporates an annular reservoir of water around and above the containment which is used as an ultimate heat sink during accident scenarios. The location and depth of this water would serve to provide a degree of shielding in addition to its primary safety function. In determining the dose under NOs, the effects of radiation scattering from adjacent surfaces and skyshine (air scatter) will be considered.

In addition to radiation exposure at the site boundary from direct radiation, effluents released during plant operation contribute to total exposures to MoP. The gaseous and liquid effluents are treated on-site to reduce activity discharged, thereby minimising the dose received by MoP. The radiation dose estimates to MoP from discharges of radioactive effluents are presented in PER Chapter 3 – RIA [4] and documented in the BAT approach [100]. Total doses to MoP from all site activities will ultimately be summarised in this chapter.

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10.6.4 Area Designation, Radiation and Contamination Zoning

10.6.4.1 SMR-300 Zoning Scheme

To meet US regulatory requirements in 10 CFR Part 20 [50], radiation zoning systems are established to classify Radiologically Controlled Areas (RCAs) and non-RCAs according to anticipated personnel occupancy and access restrictions in all areas of the facility during normal conditions.

The zoning system is used to implement radiological protection controls and to manage movement of plant, personnel or equipment. The control of such movements is a contribution towards maintaining radiation exposure ALARA.

A zoning scheme for the SMR-300 has not yet been finalised. Discussion within this subchapter is therefore based around a typical scheme in the US which the SMR-300 scheme is likely to resemble. Note the zoning scheme derived for the SMR-160 and presented in [39] will not be used as the basis for the SMR-300 design. However, as plant systems, such as HVAC, for the SMR-160 design are based on this scheme there is the potential for discrepancies within the PSR in the discussion of zones.

A typical zoning system is presented in Table 9, based on the NuScale SMR scheme [101].

Table 9: Typical Radiation Zoning Scheme in the US

Area Designation (Zone)	Indicative[5] Dose Rates (SI)	(US)	Type	Description
Unrestricted Zones				
Zone 0	≤0.5 µSv/h	≤0.05 mRem/h	Unrestricted area	Areas of plant that can be occupationally occupied without exceeding annual dose limits for members of the public of 1 mSv/y (10 CFR 20 1301 (a)) [50]
Controlled Zones (comparable to a Supervised Area under IRR17)				
Zone I	0.5–2.5 µSv/h	0.05-0.25 mRem/h	Controlled Area – limited occupancy	Areas of plant that can be occupationally occupied without exceeding personnel radiation monitoring requirements (5 mSv)[6] (10 CFR 20 1502 (a) (1)) [50]
Restricted Zones (comparable to a Controlled Area under IRR17)				
Zone II	2.5-25 µSv/h	0.25-2.5 mRem/h	Controlled Area	Areas of plant that can be occupationally occupied without exceeding the annual occupational dose limit (50 mSv). Personnel monitoring required (10 CFR 20 1201 (a)) [50]
Zone III	25-50 µSv/h	2.5-5 mRem/h	Controlled Area – limited occupancy	Areas of plant that require limited access to ensure compliance with the annual occupational dose limit (50 mSv). Personnel monitoring required (10 CFR 20 1301 (a)) [50]

[5] Ambient, whole body dose rates in these areas.

[6] 10% of annual occupational dose limit presented in 10CFR 20.1201 a [50]

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Area Designation (Zone)	Indicative[5] Dose Rates (SI)	(US)	Type	Description
Zone IV	0.05-1 mSv/h	5-100 mRem/h	Radiation Area	Areas of plant that require posting as radiation areas (10 CFR 20 1902 (a)). [50] Personnel monitoring required
Zone V	>1 mSv/h ≤10 mGy/h	>100 mRem/h ≤1 Rad/h	Restricted access High radiation area	Areas of plant that require the use of control devices to provide controlled access (10 CFR 20 1601) [50] and posting as high radiation areas (10 CFR 20 1902 (b)). [50] Personnel monitoring required
Zone VI	0.01-5 Gy/h	1-500 Rad/h	Restricted access Locked high radiation area	Areas of plant require the use of control devices to provide controlled access (10 CFR 20 16010 [50] and posting as high radiation areas (10 CFR 20 1902 (b)), [50] these plant areas will be locked
Zone VII	≥5 Gy/h	≥500 Rad/h	Restricted access Very high radiation area	Areas of plant that require controlled access with additional measures to prevent unauthorised access (10 CFR 20 1602) [50] and posting as very high radiation areas (10 CFR 20 1902 (c)) [50]

Zones will be established in conjunction with plant layout such that personnel travel times through RCAs are minimised. Additionally, to minimise the spread of radiological contamination, the layout of the zones is designed such that personnel do not have to pass through higher radiation zones to access lower radiation zones.

Access and movement between zones will be controlled with a system of engineering (i.e., physical barriers including permanent shielding, doors, locks, alarms, or combinations thereof) and administrative controls (i.e., procedures or personnel access approval).

Temporary zoning may be established during maintenance outages. These temporary zones are removed upon completion of the scheduled maintenance, following health physics monitoring.

10.6.4.2 UK Regulatory Context

The requirement for the designation of areas as either controlled or supervised is specified in Regulation 17 (1,2 and 3) of IRR17 [47]. Regulation 17 states:

1. *Every employer must designate as a controlled area any area under its control which has been identified by an assessment made by that employer as an area in which –*
    a. *it is necessary for any person who enters or works in the area to follow special procedures designed to restrict significant exposure to ionising radiation in that area or prevent or limit the probability and magnitude of radiation accidents or their effects; or*
    b. *any person working in the area is likely to receive an effective dose greater than 6 mSv a year or an equivalent dose greater than 15 mSv a year for the lens of the eye or greater than 150 mSv a year for the skin or the extremities.*
2. *An employer must not intentionally create in any area conditions which would require that area to be designated as a controlled area unless that area is for the time being under the control of that employer.*
3. *An employer must designate as a supervised area any area under its control, not being an area designated as a controlled area:*

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a. *where it is necessary to keep the conditions of the area under review to determine whether the area should be designated as a controlled area; or*
b. *in which any person is likely to receive an effective dose greater than 1 mSv a year or an equivalent dose greater than 5 mSv a year for the lens of the eye or greater than 50 mSv a year for the skin or the extremities.*

Industry best practice puts the onus on the site operator to further segregate these areas to limit exposures to personnel, either through limiting access, improving physical barriers and controls between the source and potential receptors, or implementing special procedures.

The purpose of zoning of areas is to introduce managed controls in designated areas to prevent the spread of contamination, allow for relevant PPE and RPE to be utilised, and in the case of external radiation hazards to restrict access and limit exposures. Designation of areas must be set based on the highest potential, which includes for in the event of a fault or accident, although the designed mitigations can be taken into consideration. At this current design stage, insufficient fault analysis has been completed to be included in a zoning map of the facility.

Absolute physical containment is impractical for many areas of a facility because of the need to provide for personnel access to the plant and areas within, in addition to access for services, mechanical equipment and waste products. This leads to a design philosophy where the contained material or process is surrounded by barriers. The requirement for access leads to further penetrations and further barriers enclosing zones of progressively lower contamination and external radiation potential until the external environment is reached.

10.6.4.3 Current UK practice

Radiation zoning has been used on nuclear sites for many years and is classified by different organisations according to their radiological situation and needs.

The majority of UK nuclear sites assign (external) radiation (R) and contamination (C) on a 4- to 6-point scale, with each area having a dual designation according to the hazards present, e.g., R2, C3.

Radiation (R) zones are generally segregated into ‘below supervised area dose rates, ’supervised area dose rates’ and 2 to 3 stages for ‘controlled area dose rates’ for external exposures. The higher the zone then the higher the controls (shielding, access requirements, etc.) on that area, up to prohibition of access. Time, distance, and shielding are the main dose reduction measures. Engineered controls which restrict/prevent access, provide shielding and/or distance would generally be implemented.

Contamination (C) zones are defined on the level of contamination present[7] [102], although the facilities containment and ventillation may be designed for higher levels of contamination which could occur in the event of a fault [103]. Again, the lowest level zone would result in an effective dose below which it is necessary to define the area as a supervised area, where there is no potential spread of contamination. The second zone, generally, would be where contamination levels would require the area to be designated supervised, or where conditions

[7] Also, where there is the need to keep working conditions under review, where there is a possibility of the spread of radioactive contamination.

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are such that the area needs to be kept under review. Higher zones are designated controlled areas which require engineered (e.g., containment and ventilation) and procedural (e.g., exit monitoring, training and local rules) controls, and, as a last resort, PPE to limit operator intakes and prevent the spread of contamination.

It is important to note that radiation and contamination zoning is determined by the highest potential dose that may arise for a particular operation / in a particular area. This is particularly essential for contamination zones, as the zoning is integrated with the containment and ventilation philosophy for the facility. It is crucial to ensure that an area is ventilated according to the potential contamination levels that could be present, even if these levels only occur infrequently. This would also be the case for faults with a higher probability of occurrence.

However, it is common practice to ventilate and provide the means for access control that meet the (infrequent) higher category but allows less constrained access for extended periods. For example, a zone designated C3 may be accessed for extended periods as C2, provided that ongoing monitoring shows that the de-designation (or re-designation) is appropriate. Specific risk assessments are required to get the balance right.

Using this classification system, it is normal to move from any zone to its direct neighbour, e.g., C2 to C3; however, a single structural barrier may provide the boundary between areas of two or more classes apart; e.g., C2 and C5. This would normally only be allowed where there is no direct personnel access or posting facilities across that boundary. In this way the overall retention, and therefore the activity arising in the area of lower contamination potential, is kept within an acceptable limit.

10.6.4.4 Proposed UK scheme for Generic SMR-300

To maintain compliance with the UK regulatory framework, and ensure OSW doses are ALARP, External Radiation Controlled Areas (ERCAs) and Contamination Controlled Areas (CCAs) will be established for the UK Generic SMR-300. Note the use of the term ‘ERCA’ to differentiate from the US definition of an RCA which was incorporated into the SMR-160 design. E RCAs and CCAs are kept as small as reasonably achievable, consistent with accessibility for accomplishing the tasks that must be performed in those areas.

A radiation zoning scheme is to be established to classify ERCAs and CCAs according to anticipated personnel occupancy and access restrictions in all areas of the station during normal conditions. A potential zoning scheme for the UK for the Generic SMR-300 is presented in Table 10 for the ERCA scheme and Table 11 the CCA scheme. These schemes are based upon existing UK zoning schemes and reflect the typical US scheme presented in Table 9.

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Table 10: External Radiation Controlled Area Zones and US Comparison

Area Designation	Indicative[8] Dose Rates	Max Occupancy		Description	Equiv. Zone in Table 9
		BSO (1mSv)	Target (10 mSv)		
Undesignated R0 (white)	< 0.5 µSv/h	Unlimited	Unlimited	Consideration needs to be given to expected occupancy rates to ensure exposures to personnel who do not normally work with ionising radiation are <1 mSv/y and the requirements for dose limits and targets to members of the public are met. Typical areas include offices and ancillary buildings.	Zone 0
Radiation Supervised Area R1 (green)	0.5 µSv/h to 2.5 µSv/h	400h	Unlimited	Any person, having regard to the time that could reasonably be spent in any part of that Area, is likely to receive an effective dose > 1 mSv/y 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.	Zone I
Radiation Controlled Area R2 (yellow)	2.5 µSv/h to 25 µSv/h	40h	400h	Having regard to the time that could reasonably be spent in any part of the area persons are unlikely to exceed an effective dose of 1 mSv in a calendar month. Also, an area where any person, having regard to the time that could reasonably be spent in any part of that area is likely to receive a dose greater than 6 mSv a year. Areas designated R2 may have frequent, short-term occupancy or occasional longer-term occupancy, access will be via barrier control	Zone II
Radiation Controlled Area R3 (orange)	25 µSv/h to 50 µSv/h	20h	200h	Having regard to the time that could reasonably be spent in any part of the area, persons are unlikely to exceed a dose of 1 mSv in a week. Access to R3 areas will be strictly controlled. Occupancy will be infrequent, for short time periods under written systems of work.	Zone III
Radiation Controlled Area R4 (red)	50 µSv/h to 500 µSv/h	Routine access restricted		An area where the radiation dose rate is greater than that for a Radiation Controlled Area R3 or which is subject to high dose rates (permanent, temporary, or transient). Access to R4 areas will be restricted, under a written system of work and strict dose uptake controls to limit exposures.	Zone IV
Radiation Controlled Area R5 (black)	>500 µSv/h	Routine access prohibited		An area where the radiation dose rate is greater than that for a Radiation Controlled Area R4 or which is subject to very high dose rates (permanent, temporary, or transient). Access to R5 areas will be strictly prohibited unless for safety critical purposes	Zone IV/V/VI/VII

[8] Ambient, whole body dose rates in these areas. Note that some high occupancy areas may be required to be designated controlled areas at dose rates lower than those listed here.

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Table 11: Contamination Controlled Area Zoning

Area Designation	Indicative Activity Levels [9]	Description
Non designated C0	< C1	Areas where radioactive contamination is not anticipated to occur, for example offices and areas outside of the main change room.
Contamination Supervised Area C1	Alpha emitters > 0.04 Bq/cm^2 Radionuclides not otherwise specified (including Tritium) > 0.4 Bq/cm^2	An area where radiological conditions will be kept under review; for example, the clean side of a sub-change room.
Contamination Controlled Area C2	Alpha emitters > 0.4 Bq/cm^2 Radionuclides not otherwise specified (including Tritium) > 4 Bq/cm^2	Having regard to the time that could reasonably be spent in any part of the Area, and the activities which could be performed, persons are unlikely to receive an internal exposure of 1 mSv in a calendar year.
Contamination Controlled Area C3	Alpha N/A Beta/gamma > 10 Bq/m^3	Where alpha emitting radionuclides may be disregarded
	Alpha > 0.01 Bq/m^3 Beta/gamma > 2 Bq/m^3	Where alpha and beta/gamma emitting radionuclides are present
	Alpha N/A Beta > 1.0E+4 Bq/m^3	Where tritium is present, and alpha and other beta/gamma emitting radionuclides may be disregarded
Contamination Controlled Area C4	>100 x the lower level for C3	Areas where access is generally prohibited, **For example inside vent plant.**

10.6.5 Material Selection

There are three key materials which become activated from the neutron flux in the core:

1. Material components of the core and primary circuit itself.
2. Primary coolant constituents which are transported through the core.
3. Ancillary components and structures close to the reactor vessel which are subject to neutron fluxes capable of inducing activation.

[9] Selection of values informed by ISO 17873:2004 ‘Nuclear facilities - Criteria for the design and operation of ventilation systems for nuclear installations other than nuclear reactors’ [108] and ISO 26802:2010 ‘Nuclear facilities - Criteria for the design and operation of containment and ventilation systems for nuclear reactors’ [109]

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Methods to minimise source terms resulting from neutron activation include the heavy neutron reflector which minimises the escape of neutrons and the use of alloys that contain low levels of cobalt, which when activated can lead to production of ^{60}Co, which emits relatively high energy gamma radiation. Consideration should be given to numerous other elements that could become activated and pose a significant dose or waste handling/disposal-related hazard (e.g., europium with concrete structures), and their concentrations within raw construction materials should be reduced SFAIRP in the design.

Selection of materials associated with the components/materials associated with the primary coolant circuit should minimise the quantities of ^{58}Co and ^{60}Co corrosion products which can constitute substantial radiation sources as impurities within the reactor coolant (and ancillary) systems. Chemical dosing specifications associated with reactor coolant should take into account potential constituent nuclides which could become activated and impact the source term.

The fingerprint of activation products in the reactor core and in primary coolant will be developed further. Ancillary components and structures will need to be considered also.

Material selection of the fuel assembly will be selected to ensure integrity and minimisation of activation products. It is imperative that product design and manufacture of fuel assemblies ensure reliability and integrity of the fuel cladding, to ensure containment is maintained during NOs, to minimise leakage SFAIRP. The SMR-300 design uses standard Framatome fuel for which there is a large amount of OPEX demonstrating high integrity of the fuel and cladding. See also Chapter B2 [9] for further details on the properties and performance of the fuel, and the testing and inspection carried out. Defective fuel cladding can lead to fission products being leaked into the primary coolant thus increasing the associated source term, which can also impact upon gaseous and aqueous discharges to the environment.

The design will be developed further to ensure minimisation of activation products in the core and coolant, as discussed in Chapter B23 [24].

10.6.5.1 Chemistry control

Water chemistry has a significant impact on the source term. The chemistry regime acts to minimise corrosion of the primary structures (thereby minimising source term) and supports the ageing management of SSCs, minimising maintenance and the associated doses incurred from both the source term and (potentially) as a result of reduced periodic radiography inspections to assess SSC integrity.

The optimisation of the reactor design and its associated systems (including the SFP) to control impurities and activation products, such that the risks are reduced to ALARP is described in Chapter B23 [24]. SFP, reactor chemistry RGP and chemistry guidelines are also discussed in Chapter B23 [24]. Optimum radiochemistry control parameters associated with chemistry regime will be developed with defined limits, to ensure the integrity of safety barriers and radiological safety. Option evaluations of risks affecting Reactor Chemistry are presented in Chapter B23 [24], along with the justification as to why/how these risks have been reduced to ALARP.

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The RIA in PER Chapter 3 [4] and the planned Normal Operations Dose Assessment (NODA) will feed back into the chemistry chapter B23 [24] to demonstration risk reduction has taken place.

10.6.5.2 System and Equipment Design

During in-service plant operation, plant maintenance and inspection activities inside the reactor island are anticipated to contribute a significant proportion of the collective worker dose. Key aspects to system and equipment design are to utilise equipment that does not require maintenance, requires infrequent or minimal maintenance and the facility is designed to allow for easy access and for the use of protective measures. Equipment to be avoided would be plant items that require frequent maintenance or breaches of containment.

The design life of the Plant is 80 years. Due to cumulative build-up of the source term, along with the potential for an increasing frequency of outages and non-routine maintenance, it is critical the design features are developed during the design phase to consider dose exposure to workers over the whole operational lifecycle. The SMR-300 has a number of typical design features which are conducive to ensuring doses to OSWs and MoP have been reduced ALARP:

- The reactor vessel design includes an integrated head package which combines the head lifting rig, Control Rod Drive Mechanism (CRDM), lift columns, CRDM cooling system, and instrumentation cabling into an effective, one-package reactor vessel head design. Mounted directly on the reactor vessel head assembly, the system helps to minimise the time, manpower, and radiation exposure associated with head removal and replacement during refuelling operations.
- The reactor design also includes a heavy reflector to minimise neutron leakage and reduce activation and embrittlement of the core vessel.
- Integral to the design is permanent shielding for reducing working area dose rates from the CRDM drive shafts. The combination thermocouple / in-core detector system is not kept with the head assembly during refuelling, but instead remains with the upper reactor vessel internal components. This allows the thermocouple / in-core detector system to be shielded underwater in the refuelling cavity during most refuelling operations, reducing dose rates around the head assembly. The reactor vessel nozzle welds are designed to accommodate remote inspection with ultrasonic sensors, removing the need for in person inspection of the welds.
- The design shall provide provisions for draining, flushing, and if necessary, remote cleaning or decontamination of equipment containing radioactive material. Drainage shall be directed to a sump or a closed system.
- Installed in the ventilation systems are HEPA filters designed for the capture of airborne radiological material. Filter removal involves use of semi-remote tools. The design incorporates adequate space around the filter locations to remove the filters for transport to a storage and packaging area. The replacement process is designed to minimise exposure and inadvertent release of radioactivity to the environment. The filter handling tools are of an uncomplicated design to minimise susceptibility to malfunction.
- Demineralisers in radioactive wastewater systems are designed so that the spent resins can be remotely and hydraulically transferred to spent resin tanks. Fresh resin is loaded into the demineraliser remotely. The demineralisers and piping can be

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flushed with demineralised water. The system design prevents inadvertent flushing of the resin into the purification loop.

- Tanks containing radioactive liquid are provided with sloped bottoms and bottom outlet connections. Overflow lines are directed to the waste collection system to control contamination within plant structures. Tanks containing radioactive liquid are fabricated from or lined with stainless steel. Sumps which can contain radioactive liquid are lined with stainless steel to facilitate decontamination and reduce build-up of radioactive material.
- Heat exchangers are designed to facilitate tube and other associated inspections. Heat exchangers are provided with corrosion-resistant tubes to reduce leakage. Heat exchanger filters are shielded to adequately reduce the external dose rate originating from the trapped contaminants. Impingement plates are provided and both tube side and shell side flow velocities are limited to minimise erosive effects. Wherever practicable, the radioactive fluid passes through the tube side of the heat exchanger.
- Instrumentation requiring regular monitoring or access will be installed in a low radiation area whenever practicable [39]. Instruments located in a high radiation area can be removed and taken to low radiation areas for calibration.
- Instrumentation line connections are located above the pipe midline to minimise the build-up of deposits, whenever practicable.
- Motor-operated, air-operated, or other remotely actuated valves are used to minimise personnel exposure from valve operations. Valves that require frequent manual operation are shielded or have extenders to minimise proximity to high radiation areas during operation. Valves in high radiation areas that require infrequent operations may have remote manual operators. Valve operation of certain valves in high radiation areas may be limited to times when the facility is shutdown to reduce exposure to the plant personnel.
- Piping is designed to minimise deposit traps using bends rather than elbows. Welds are applied in a manner to prevent deposit traps from forming. Piping is routed away from normal egress areas or is shielded to keep exposures to a minimum.
- Whenever practical, pipes containing radioactive substances are separated from those that do not contain radioactive substances. Pipes containing RM are labelled to indicate the radiation hazard.
- Containment penetrations are minimised and designed with shielding to minimise streaming through containment from radiation sources (see also 10.6.3).
- Equipment vents and drains are piped directly to a collection device connected to the collection system. Immediate collection of the air vented by equipment minimises the airborne contamination. Additionally, preventing potentially radiologically contaminated liquid from flowing across the plant floor reduces the potential for evaporation or off-gassing of the RM.
- Welded piping systems are employed on systems containing radioactive fluids to the maximum extent practicable. If welded piping systems are not employed, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the collection system.
- Coatings are applied to the concrete floors and walls of areas which could potentially become contaminated, to facilitate decontamination.
- Design of equipment incorporates features that minimise the spread of radioactivity during maintenance operations. These features include flush and drain connections on pump casings for draining and flushing the pump prior to maintenance and flush connections on piping systems that could become internally contaminated.

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- The SFP for SMR-300 is located inside the containment structure. Water in the SFP shall be maintained at a low activity level by means of a purification system with particulate filters and ion exchangers. For any failed fuel removed from the reactor core, the design shall provide a means to prevent any solid radioactive material from contaminating the pool water. Failed fuel will be placed into a damaged-fuel container in designated cells in a spent fuel rack, which allows water to pass over the fuel for cooling but includes filtration to capture any solid fines and prevent their dispersal in the SFP water.
- The selection of construction and shielding materials and the method of construction will aim to reduce the generation of activation products within these structures, through for example, the use of low-europium concretes, low-cobalt steels, and positioning of rebar within concrete structures.

The dose assessment of the SMR-300 will take these design aspects into consideration when defining further risk reduction measures for the Generic SMR-300 to demonstrate ALARP.

10.6.6 Layout design

In line with HoCMs, the design should utilise plant layout design features SFAIRP before other control measures such as systems of work, use of personal protection equipment and respiratory protective equipment are adopted. Wherever possible, the reliance on engineered access control shall always be preferred over administrative access control. This is an expectation of the design that will be assessed.

ERCAs associated with higher dose rates should be minimised, with efficient accessibility for accomplishing the services that must be performed in those areas, including equipment laydown requirements. ERCAs where employees spend greater time should be designed to lowest practical dose rates.

To minimise radiation doses to personnel working in the ERCAs and the spread of contamination from/between CCAs, the layout of the ERCAs and CCAs shall be designed such that personnel do not have to pass through higher radiation/contamination zones to access lower radiation/contamination zones e.g., R2 to R1.

ERCAs contain radioactive material and equipment, and the presence of contamination is to be presumed. Where contamination is expected to be present the emphasis should be on containment as part of Plant design, therefore any sources of contamination in the area are localised and under control. The movement of radioactive material within the area could temporarily create contamination, which will be cleaned up on completion of the activity, therefore areas that could become contaminated should be designed with easily de-contaminable surfaces. The IAEA Manual on Decontamination methodologies and approaches [104] and IAEA TECDOC on lessons learnt [105] will be used to support assessment of the suitability of the design.

The zoning system described in sub-chapter 10.6.4.4 shall be used to inform the layout design to minimise personnel exposures. The following considerations are to be made when designing the layout of the facility to reduce doses to workers to ALARP:

- A system shall be established to control movements between the zones. This system will include engineered features (i.e., physical barriers including permanent shielding, doors, locks, alarms, or combinations of these alternatives) and administrative controls.

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- Movements between adjacent zones shall be controlled by a combination of physical barriers (e.g., walls and doors), contamination controls, and procedural controls.
- Access control barriers shall be designed to permit emergency egress from an access-controlled area without the use of tools or external help.
- Access controls to R4 areas shall be achieved by the provision of lockable doors and, where appropriate, the use of interlocks to ensure that access is only permitted when radiation levels are acceptably low, or appropriate systems of work are in place to allow authorised access.
- Overrides shall be prevented, except by deliberate operator override action. Suitable, installed radiation monitors shall be provided in areas with relatively high external radiation and/or contamination levels.

The above should ensure that access and egress (especially emergency egress) is optimised through the arrangement of plant items to better co-locate zones, whilst minimising the need to pass through areas with higher levels of dose or contamination.

Changeroom design should follow RGP as documented in Changeroom Design, Operation and Maintenance, A Nuclear Industry Code of Practice [106] to ensure risks are reduced to ALARP.

10.6.7 Ventilation design

The SMR-300 includes a number of ventilation systems designed to control and minimise the airborne contamination and gaseous radiation sources present during the operation of the plant. These HVAC Systems will be designed such that separate systems will service areas depending upon their contamination classification and locations within the buildings/installations defined in 10.4.2.1. The HVAC System is described in PSR Chapter B5 [10].

Protection from airborne radioactive material is achieved through the following:

- 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.
- Ventilation systems and the Gaseous Radwaste System (GRW) are designed to contain radioactive material that has been deposited, collected, sorted, or transported within or by the systems, such as those associated with the coolant equipment enclosures, and active waste treatment, and other isolated areas.
- Separate fan and filter rooms are provided for each C classification designated within the building. Fan and filter rooms for the undesignated plant areas and buildings are located outside of the RCA areas to limit dose uptake during filter changeout and maintenance in uncontaminated facilities.
- Task-specific and mobile ventilation systems (Mobile Filtration Units) will contain HEPA filters, and activated charcoal filters as appropriate, to control local airborne contaminants originating when equipment containing potential airborne sources is opened to the atmosphere.

The ventilation systems include design features to control the release of airborne radioactivity. These design features include:

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- In building compartments with a potential for contamination, the exhaust is designed for greater volumetric flow than is supplied to that area. This creates a depression within the area thus minimising the amount of uncontrolled exfiltration from the area.
- Consideration is given to the potential disruption of normal airflow patterns by maintenance operations, and provisions are made in the design to prevent adverse airflow direction.
- Air discharged from the containment is passed through HEPA filters and charcoal adsorbing filters to remove airborne radioactivity. Air exhausted from the auxiliary building, fuel handling area of the auxiliary building, and the annex building is monitored for radioactivity. Means are provided to shut off supply air and divert exhaust air through HEPA filters and charcoal adsorbers upon detection of airborne activity in excess of prescribed levels.
- Alarms are provided in the main control room for these discharge flows, and for flows from the RWB and the Health Physics (HP) /hot machine shop area. These alarms alert the operator of high radioactivity concentrations in the air. Prompt response to Alarms and addressing the cause minimises the discharge of contaminants to the environment and reduces in-plant exposures.
- Tanks or vessels which contain radioactive materials are vented to the respective building ventilation system for release to the monitored plant vent.

During outages and other maintenance activities the use of Local Extraction Units containing HEPA filters can be utilised where airborne radiological hazards have been identified as a result of risk assessments.

Design of ventilation systems for the Generic SMR-300 should consider the following UK RGP: Ventilation systems for radiological facilities, design guide [107], ISO 17873:2004 [108], and ISO 26802:2010 [109], implementation of these guides will aid in the demonstration of reducing doses to ALARP.

A review of the SMR-300 ventilation system design, against UK RGP in conjunction with the proposed UK area designation scheme in subchapter 10.6.4.4 will be presented in the IRR 2017, SAPs and TAGs Compliance Matrix for the Generic SMR-300.

10.6.8 Radiation and contamination monitoring

Instrumentation required for radiation monitoring for the protection of OSWs will be captured within an in-plant program of monitoring and air sampling. Monitoring instruments will be located throughout the Generic SMR-300 plant to provide measurements of direct radiation fields and concentrations of airborne radionuclides, as described in System Design Description for Radiation Monitoring System [110]. The radiation monitoring program will use both fixed and portable instrumentation.

Stationary alarming dose-rate meters will be provided:

- For monitoring the local radiation dose rate and airborne contamination levels at places routinely occupied by operating personnel and where changes in radiation levels are such that access may be limited for periods of time (across radiation zones and non-ERCA/CCA interface boundaries).
- To indicate the general radiation level at appropriate locations in the event of DBAs and, as far as practicable, severe accidents.

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- To give sufficient information in the control room or at the appropriate control position to enable plant personnel to initiate corrective actions when necessary.

Provision of suitable power outlets and space allocations for fixed radiation/contamination monitoring stations shall be made at interfaces between the ERCAs and non-ERCAs, and CCAs and non-CCAs.

Radiation monitoring of process systems is provided in the Radiation Monitoring System (RMS):

a) To detect the failure of a confinement barrier; and
b) To detect leakage of radioactive material from or into a process system.

These systems are discussed for the SMR-160 in SMR-160 Radiation Monitoring System Detector Specification [111] and SMR-160 Radiation Monitoring System Design Requirements Document [112].

Radiation monitors can be used as part of an interlocked system to ensure personnel cannot enter areas of significant radiological hazards. A radiation and contamination monitoring program will be developed in the detailed design stage of development. Radiation metrology and personal dosimetry for use within the generic SMR-300 design will be discussed in a regulatory compliance matrix.

10.6.9 Criticality Controls

An inherently safe design is one that avoids radiological hazards rather than controlling them. It prevents a specific harm occurring by using an approach, design or arrangement which ensures that the harm cannot happen, for example a criticality safe vessel. The HoC for radiological protection (Table 8) and the ERIC/PD philosophy for hazard reduction (subchapter 10.6) are also pertinent to criticality safety and give preference to minimising the amount of fissile material present, consistent with the process requirements.

The principal means of passive engineering control of criticality should be geometrical constraint. Where sub-criticality cannot be maintained through geometrical constraint alone, additional engineered safety measures should be provided, such as fixed neutron absorbers. Assessment of the adequacy of designed criticality controls is yet to be performed. Additional design work will be carried out as necessary following the outcome of these assessments. Further safety measures may need to be provided, for example to:

a) Control the mass and isotopic composition of the fissile material present.
b) Control the concentration of fissile material in solutions.
c) Control the amount of neutron moderating and reflecting material associated with the fissile material.

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The design and operation of plant and equipment dealing with fissile material should facilitate the termination of a criticality incident.

UK experience and application of RGP for fuel burnup credit (typically assumed for criticality calculations supporting the safety case) is limited due to there being only one commercial PWR in operation. Therefore, the methods for demonstrating criticality control will be developed based on demonstration of geometric design, and other engineered design features.

At this preliminary stage of the design, the focus for accident analysis via deterministic, probabilistic, and severe accident analysis is on demonstrating that off-site consequences can be reduced to ALARP. Chapters B14 [6], B15 [17] and B16 [18] will be developed during the GDA and a Preliminary Fault Schedule (PFS) presented. A review of where access controls, shielding, other engineered controls, design features and safety features are claimed to limit exposures to OSWs and MoP will be conducted and presented in the next revision of Chapter B10. The criticality controls for all out-of-core systems handling fissile material will be reviewed to ensure risks of a criticality event are reduced to ALARP. As part of the radiological protection review, safety considerations will be addressed by reference to relevant SAPs [70], e.g., on criticality, and ALARP (including TAG 041 on criticality safety [78] and TAG 005 ALARP [74]). Relevant IAEA [113] and Nuclear Energy Agency (NEA) guidance [114] on use of burnup credit in safety cases will also be considered.

During the site-specific phase, detailed criticality assessments for fuel and waste handling will determine the clearance requirements to be adopted during the commissioning and operational phases when fuel is on site.

10.6.10 Review of Radiation Protection Claimed in Accident Protection

ALARP in principle is demonstrated by following the HoC for NOs (Table 8). Together with the ALARP in design practice of following the ERIC/PD philosophy for hazard reduction (subchapter 10.6) it gives a design for radiological protection that is likely to demonstrate that radiological exposures have been minimised and that risks from this have been reduced to ALARP. This will satisfy Claim 2.1.1 and Claim 2.2.3.4, respectively.

To demonstrate that the ERIC/PD philosophy has been followed, Chapters B14 ‘Safety and Design Basis Accident Analysis’ [6], Chapter B15 ‘Beyond Design Basis, Severe Accident Analysis, and Emergency Preparedness’ [17] and Chapter B16 ‘Probabilistic Safety Analysis’ [18] provide a safety analysis to identify hazards that need to be reduced. Safety functions for SSCs (including radiological protection SSCs) for hazard reduction are derived here.

- Chapters B14 [6] presents the deterministic analysis for the SMR-300 following accident conditions and presents the basis for demonstration that the risk is ALARP in comparison with the NT for design basis accidents introduced in Chapter A2 [8]. This will include those internal and external hazards identified in Chapters B21 ‘External Hazards’ [15] and B22 ‘Internal Hazards’ [16].
- Chapter B15 ‘Beyond Design Basis, Severe Accident Analysis, and Emergency Preparedness’ [17] provides an assessment of the SMR-300 following accident conditions that are low in frequency or high in consequence.
- Chapter B16 ‘Probabilistic Safety Analysis’ [18] presents the probabilistic analysis for the SMR-300 following accident conditions to meet probabilistic risk criteria introduced in Chapter A2 [8].

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At this preliminary stage of the design, the focus for accident analysis via deterministic, probabilistic, and severe accident analysis is on demonstrating that off-site consequences can be reduced to ALARP. Chapters B14 [6], B15 [17] and B16 [18] will be developed during the GDA and a PFS presented. A review of where access controls, shielding, other engineered controls, design features and safety features are claimed to limit exposures to OSWs and MoP will be conducted and presented in the next revision of Chapter B10.

10.6.11 CAE Summary

The design for radiological protection for the SMR-300 already considers extensive dose reduction measures to meet the ALARA principle and the identified radiological protection requirements in the US. Significant developments have taken place during the design in terms of source term minimisation, material selection and chemistry control to meet Claim 2.2.3.4.3, system and equipment design and layout design to meet Claim 2.2.3.4.5, ventilation design to meet Claim 2.2.3.4.4, and radiation and contamination monitoring in support of Claim 2.2.3.4.6.

Future developments for the generic SMR-300 design include shielding design optimisation to meet Claim 2.2.3.4.2, area designation to meet Claim 2.2.3.4.6, reviews of criticality controls to meet Claim 2.2.3.4.1 and radiological protection in accident analysis in support of Claim 2.1.1.3. These developments help to demonstrate ALARP in design practice and the control of radiation exposure and the control of the release of radioactive material, such that Claim 2.1.1.3 can be met.

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10.7 DOSE ASSESSMENTS FOR WORKERS AND MEMBERS OF THE PUBLIC

Claim 2.1.1.2: Effective doses to workers and members of the public during normal operations are assessed.

Considering the radiological protection requirements and the design measures for radiological protection identified above, this subchapter sets out how doses are assessed to meet the limits and targets and demonstrate that risks are reduced to ALARP.

10.7.1 Sources of Radiation

The NO radiation sources vary for the different plant states, and can be sub-divided into three broad categories:

- Operations where the reactor is at power, including start-up, shutting down, normal full-power operation and abnormal operations within the design basis.
- Operations that are conducted when the reactor is shutdown, including refuelling operations (inside containment) and during outages for maintenance.
- Operations outside of the reactor that can be conducted either at power or shutdown, including waste handling operations, fuel handling operations (outside containment), in-service maintenance and inspections.

As explained in subchapter 10.6.2, source terms are defined for NO and accident conditions as a design tool to aid the reduction of exposure and radiological risk. Development of the detailed source terms with regard to radionuclide fingerprint and quantities will inform the detailed design requirements.

The following subchapters address each category of plant operation in turn.

10.7.1.1 Radiation Sources for At-Power Operation

Although not an exhaustive list, radiation sources (including fixed and non-fixed radioactive materials) are generated when at power in the Reactor Core (RC) through the following key routes:

1. Fission products leakage from fuel defects which can become mobile in the reactor coolant.
2. Neutron and Gamma radiation leakage direct from the core as a result of the fission process.
3. Activation of material wear and corrosion products that are circulated in the reactor coolant and activation of coolant itself. Process piping leading out of the core containing reactor coolant.
4. Radiation sources in the CVC system consist of radionuclides carried in reactor coolant. The chemical and volume control system components used for purification of reactor coolant are located in the RAB.
5. The SFP and cooling system to clean the water in the SPF will be a source of activity accumulation on filters (including filter media such as resins) and deposits in the bottom of the pool. In certain conditions this may lead to gaseous and alpha emissions.
6. Leakage of the primary coolant to the secondary system can take radioactivity outside of containment.

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10.7.1.2 Radiation Sources for Shutdown Operations

One of the largest contributors of personnel annual exposure will be as a result of shutdown maintenance and refuelling operations. In this instance the source terms will be different and typical examples are listed below:

1. During shutdown of the reactor, radiation emissions will reduce significantly; however, the radionuclides formed by fission and activation will remain.
2. During shutdowns/outages refuelling will take place which leads to the potential exposure related to handling, storage, and cross-site transfer of fuel and redundant in-core components. Spent fuel gamma and neutron source strengths are used to determine or inform the necessary control measures.
3. Residual heat removal (RHR) system and other systems put into service during outage and containing RCS fluid. The estimates for source strength from the RHR system are based on reactor coolant at normal full power operation.

Maintenance of activated primary and auxiliary components located above the reactor core and only accessible during shutdown (e.g., control rods and drive mechanisms) are a source of radioactivity with the predominant activation product in terms of dose being Co-60.

10.7.1.3 Radiation Sources outside of Reactor Operations

Sources that could be accessed during both at-power and shutdown operations are:

1. Radioactivity in the LRW systems includes contribution from fission and activation products in the reactor coolant as well as a result of plant operations. Radiation sources for the various pumps in the liquid radioactive waste system are assumed to be identical to the liquid sources in the tank.
2. Radioactivity in the GRW Systems including the HVAC systems. Radiation sources include filtration units and ducting.
3. Spent fuel and redundant in-core components, including during cross-site transport and when stored in the ISFSI.
4. The SRW System handles various radioactive waste products ranging from low to high activity. Low activity materials include contaminated item such as protective clothing, papers, and air filters. Higher activity sources, such as spent resin, are processed in a separate program to the low activity materials. Solid radioactive waste remains in on-site storage until processed or dispatched for offsite treatment.

Radioactive function check and calibration sources are associated with activities to support NOs of the generic SMR-300, such as sources used for calibration of instrumentation.

The EIMT schedule is likely to include inspections using radiography sources. This is deemed out of scope for this assessment as the selection of inspection equipment is considered to be operations rather than design, however, the need for inspections will be reflected within the design.

10.7.2 Sources of Contamination

During NOs (including outages) at power or shutdown there are many ways that sources of radiation can become mobile and present a radiological contamination hazard in the form of an airborne Radioactive Material (RM) or loose surface contamination. The majority of airborne RM within the plant results from equipment leakage.

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Typical areas and tasks where airborne activity occurs include:

1. Containment Atmosphere - Personnel access to containment is not permitted during power operations. Prior to reactor shutdown the air in the containment will be purged. Initial entry after shutdown will consist of operator and HP personnel in Self-Contained Breathing Apparatus to ensure the atmosphere, dose rates and airborne levels of contamination are as expected, and containment is suitable for entry. When personnel are present in containment, during non-power operations, additional purging will be performed to reduce personnel exposure. The airborne radiological contamination hazard includes noble gases, particulates, and radioiodine.
2. Auxiliary Building Atmosphere - The primary source of airborne activity in the auxiliary building is from equipment leakage.
3. Maintenance activities where it is necessary to breach containment structures for repairs to plant items, maintenance of seals, or changing out of filters, etc.

Contaminated surfaces are more readily encountered during outages, and generally occur when containment has been breached for maintenance activities. Examples of areas where surface contamination can occur include:

1. Crud deposits in the reactor coolant system, especially in crud trap areas encountered during maintenance tasks.
2. Handling of spent filters and filter media, from liquid and gaseous systems.
3. Handling of plant items removed for maintenance or disposal.
4. Activities associated with the SFP such as transfer and handling of MPCs containing fresh fuel, spent fuel or redundant in-core components.

Contamination control is demonstrated by the design of containment and ventilation systems, and by the application of good house-keeping practices.

10.7.3 Exposure Pathways

10.7.3.1 External Radiation Exposure

Through NOs, doses to OSWs and off-site MoP are predominantly due to external radiation exposure. In addition to the type and quality of the emitted radiation, the scale of radiation exposure (leading to radiation dose uptake) is determined by:

1. **Time** in the radiation field.
2. **Distance** from the radiation source, controlled via, for example, design features or use of remote tooling.
3. **Shielding/Decay** performed to reduce the magnitude of the source of radiation, through reduced transmission or by allowing time for radioactive decay.

10.7.3.2 Internal Radiation Exposure

Even though NOs doses are dominated by potential external radiation exposure, there exists potential for intakes of RM (contamination) leading to a Committed Effective Dose (CED) following, for example, breaks in containment of process systems for maintenance. The amount of internal exposure leading to a CED is determined by:

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1. The type, physico-chemical form and extent of the contamination (activity concentration).
2. The quantity of radioactive material entering the body.
3. The time interval over which material resides in the body (i.e., some radionuclides (and chemical forms) have longer biological half-lives than others).

Principal pathways of RM contamination entering the body are via:

1. Inhalation.
2. Ingestion.
3. Absorption through skin.
4. Injection.

Control measures that can be adopted to prevent and/or limit CED from internal exposures due to RM contamination are as follows (see also Sub-Chapter 10.5.2.1):

1. Limit the generation of RM and quantity of RM handled and therefore available for potential intakes.
2. Use of engineering controls to maintain the containment of the RM within process systems etc., SFAIRP, to minimise the potential for the generation of contamination.
3. Prevention of dispersal, following any necessary loss of containment, or in the event of an inadvertent release due to a fault/accident; achieved via procedural controls (including temporary engineering controls) or contingency plans.
4. Provide a barrier between OSWs and non-fixed radioactive material, such as Personal Protective Equipment (PPE) and Respiratory Protective Equipment (RPE).

10.7.4 Worker Dose Assessment – Normal Operation

In order to complete high-level worker dose assessments prior to NOs, data on sources, shielding, occupancy, maintenance activities and EIMT scheduling are required.

Dose assessments are carried out to inform design changes to be implemented in the generic SMR-300 design to achieve design dose constraints and drive towards an ALARP design. Therefore, dose assessments must be carried out for the following types of workers to meet the dose limits and targets presented in subchapter 10.5.1:

- Target 1 provides the BSL which is also the Legal Limit (LL) and the Basic Safety Objective (BSO) for ***any person on the site***.
- Target 2 details the BSL and BSO for ***any group on the site***.

Assessment of doses to individual workers and collective doses for groups of workers are required to evaluate whether the design is ALARP.

The NODA is a key document to support the Radiological Protection Chapter B10 with UK context to demonstrate compliance with regulatory requirements, for example against ONR TAG NS-TAST-GD-043 Radiological Analysis - Normal Operation [79]. This document will assess individual doses to workers performing high exposure tasks (such as maintenance during an outage) to demonstrate that doses will be below the legal limit. It will also evaluate collective doses to workers during non-outage years, basic refuelling outage, standard refuelling and maintenance outage and major outage involving repair or replacement of a significant plant item. Assessment of direct radiation doses to members of the public from all

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major onsite sources will also be provided in the NODA – and input to the RIA within PER Chapter 3 [4].

10.7.4.1 Comparison of Radiation Doses for SMR-300 against Dose Targets

Radiation exposures to plant operating personnel are determined in accordance with the methods described in Regulatory Guide 8.19: Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage ManRem Estimates [115] and Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants [116]. Table 14 provides a draft template for the presentation style for the assessments based on RG 8.19 [115].

No values for doses/dose rates or confirmation of source terms were available for this publication; where data is available, preliminary calculations and results will be presented at the next revision. Individual and collective doses to OSW will be summarised here together with a comparison against the limits and constraints presented in subchapter 10.5.1, i.e., Targets 1 and 2, with the aim to demonstrate that doses are ALARP.

10.7.4.2 Comparison of Radiation Doses for SMR-300 against comparable PWR Facilities.

Radiation doses to OSWs for SMR-300 facilities will be compared against comparable PWR facilities on a per-GW power basis, where possible. This is to allow a comparison of working practices and assumptions, taking cognisance of the difference in reactor size between a SMR and traditional reactors.

Holtec's aim is to operate the SMR-300 with lower dose uptake than comparable PWR facilities.

10.7.5 Public Dose Assessment – Normal Operation

Assessment of dose to MoP from direct radiation from plant facilities will be assessed and presented here. The generic site description is provided in PSR Part A Chapter 2 'General Design Aspects and Site Characteristics' [96], this states that the nearest member of public will be located 100 m from a stack and sources of external dose.

Note that the spent fuel store could be located closer to the site boundary than the stack, so dose assessments will take relative location of buildings into account. Further details of the exposure groups are provided in PER Chapter 3, RIA [4]. Occupancy data for representative MoP living close to the facility are presented in Table 12, the presentation format for direct radiation doses to the representative MoP is shown in Table 13.

Table 12: MoP Occupancy Data

Parameter	Infant	Child	Adult
Occupancy at habitation (h/y)	8760	8760	8760
Fraction of time spent indoors	0.9	0.8	0.5
Indoors Shielding factor	0.1	0.1	0.1

Table 13: MoP Direct Radiation Doses – example table

Age Group	Location	Dose Rate (µSv/h)	Occupancy (h)	Annual Dose (µSv)
Adult	Indoors	TBC	4380	TBC
	Outdoors	TBC	4380	TBC

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Age Group	Location	Dose Rate (µSv/h)	Occupancy (h)	Annual Dose (µSv)
	Total	-	8760	TBC
Child	Indoors	TBC	7008	TBC
	Outdoors	TBC	1752	TBC
	Total	-	8760	TBC
Infant	Indoors	TBC	7884	TBC
	Outdoors	TBC	876	TBC
	Total	-	8760	TBC

10.7.5.1 Summary of MoP Exposures

The RIA of the generic SMR-300 twin-reactor site in PER Chapter 3 [4] assesses doses to reference MoP from gaseous and aqueous discharges. These will be summarised here following the completion of the NODA and RIA to allow comparison of estimated bounding doses against the limits and constraints presented in subchapter 10.5.1. Target 3 details the BSO and BSL for MoP and other persons off the site.

10.7.6 Future Dose Optimisation

As the design of the generic SMR-300 develops past the GDA phase and enters detailed design, further opportunities for dose reduction for the exposed groups will be realised. Radiological protection safety functions will be defined for SSCs developed during the detailed design stage.

During the commissioning phase, as radioactive sources and fuel are brought onto site, dose assessment will be fine-tuned and optimised to reduce dose rates for operations to ALARA and risks to ALARP.

10.7.7 CAE Summary

The design for radiological protection for the SMR-300 already considers extensive dose reduction measures to meet the ALARA principle and the identified radiological protection requirements.

Dose assessments will be performed for NOs for the generic SMR-300 to meet Claim 2.1.1.2. The assessed doses will be compared against the identified radiation protection requirements in the following revision of this document. There is a high degree of confidence that the dose assessments will demonstrate that limits and targets can be achieved for this stage of the design. Once this is confirmed, any remaining design decisions that can be taken to reduce dose uptake further will be considered. These risk assessments will be revisited during detailed design, commissioning and operations as more detail on the operation of the reactor becomes available, with the aim to optimise dose uptake further.

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Table 14: Annual Occupational Personnel Dose Estimates Template (based on US NRC Reg Guide 8.19 [115])

Activity	Dose rate (mSv/h)	Time per activity (h)	Dose per worker (mSv)	Actions per year	External Dose Ind. (mSv/y)	No. Workers	External Dose Col. (person-Sv/y)
Operation of Systems Equipment Routine Maintenance Inservice Inspection Special Maintenance Waste Processing Refuelling Activities Health Physics Activities Other (specify)	Dose rate from all sources present at the OSW working position when carrying out a specific activity	Time taken to complete a single instance of the specific activity	Product of dose rate and time per activity	Number of times (on average) that an activity will be completed per year. (outage tasks assumed to be completed in a single year)	Total annual external dose received for completing each activity	In support of assessment of collective dose, number of OSW involved in completing each activity in a given year.	Product of individual dose and number of workers for each activity.

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10.8 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the radiological protection Chapter and how it contributes to the overall demonstration of ALARP for the generic SMR-300. PSR Part A Chapter 5 [26] sets out the overall approach for demonstration of ALARP and how contributions from individual Chapters are consolidated.

This subchapter therefore consists of the following elements:

- Technical Summary;
- ALARP Summary
    - Review against RGP.
    - Evaluation of Risk.
    - Risk Reduction Options.
    - GDA Commitments and Forward Actions.
- Conclusion.

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

10.8.1 Technical Summary

PSR Chapter B Part 10 [117], Revision 0 identifies that the radiological protection aspects of the SMR-300 can meet the high-level Claims of the SSEC. This is demonstrated through claims focused on the identification of radiological protection requirements for NOs and that SSCs are designed to meet radiological protection requirements and minimise exposures.

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.

The dose limits and targets for the SMR-300 design have been developed to comply with the ALARA principle. Compliance with the ALARA principle is achieved by establishing design dose constraints that are lower than the limits and targets, based on OPEX and best nuclear industry practices. This degree of conservatism allows the SMR-300 to meet the US defined ALARA principle and reduces worker dose as priority.

The application of the ALARP principle and IRR17 are the main changes that will drive any design decisions for the generic SMR-300 in accordance with RGP in the UK. This includes design constraints, dose limits and targets and design principles, such as the HoCMs.

The design for radiological protection for the SMR-300 already considers extensive dose reduction measures to meet the ALARA principle. Dose assessments will be performed for NOs for the generic SMR-300. The assessed doses will be compared against the identified radiation protection requirements in the following revision of this document. There is a high degree of confidence that the dose assessments will demonstrate that limits and relevant targets can be achieved for this stage of the design. Once this is confirmed, any remaining design decisions that can be taken to reduce dose uptake further will be considered. These assessments will be

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revisited during detailed design, commissioning and operations as more detail on the operation of the reactor becomes available, with the aim to optimise dose uptake further.

Claim 2.2.3.4: SSCs are designed to meet Radiological Protection requirements and minimise exposures.

The design for radiological protection for the SMR-300 already considers extensive dose reduction measures to meet the ALARA principle and the identified radiological protection requirements in the US.

Significant developments have taken place during the design in terms of source term minimisation, material selection, chemistry control, system and equipment design, layout design, ventilation design, radiation and contamination monitoring. Future developments for the generic SMR-300 design include shielding design optimisation, area designation, and reviews of criticality controls and radiological protection in accident analysis. These developments help to demonstrate ALARP in design practice and the control of radiation exposure and the control of the release of radioactive material.

10.8.2 ALARP Summary

10.8.2.1 Review against RGP

RGP is identified in sub-chapters 10.4 and 10.4.2.1. The generic SMR-300 design is reviewed against this RGP in sub-chapters 10.4.2.1 and 10.5. Once data is available, assessments presented in sub-chapter 10.7 will demonstrate where RGP is followed and identify areas where further improvements are required.

Lessons learnt from previous GDAs have been extensively reviewed for good practice in Appendix A and Appendix B.

In terms of radiological protection, Holtec aim to demonstrate that UK RGP has been followed, that good practice and learning from experience has been applied, that risks are reduced to tolerable levels, and potential exposures of OSWs and MoP are ALARP (or are capable of being reduced to ALARP, subject to further design development).

10.8.2.2 Evaluation of Risk

The NT against which the demonstration of ALARP is considered for radiological protection aspects for NOs can be found in PSR Chapter A2 [8] and are replicated in Table 7.

At this time, the evaluation of the NOs against Targets 1-3 has not been provided. Data will be presented in a format consistent with Table 13 for MoP and Table 14 for OSWs. There is a high degree of confidence that the dose assessments will demonstrate that limits and targets can be achieved for this stage of the design, and it is expected that the effective doses will be below the relevant BSLs and tolerable. Holtec aim to then demonstrate that the risks have been reduced to ALARP in accordance with regulatory expectations.

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10.8.2.3 Risk Reduction Options

Dose reduction options have been considered throughout this chapter. As described in subchapters, a key aspect of an 'ALARP' determination is adherence to the HoC principle and ensuring that the design process undertakes all necessary steps to ensure primacy is afforded to consideration of engineering controls, design features and by the provision and use of safety features and warning devices, to restrict SFAIRP the extent to which employees and other persons are exposed to ionising radiation.

In accordance with the HoC, controls to address the radiological aspects have been identified throughout the chapter, and which are or will be incorporated within the design, in order to fulfil the statutory requirements of IRR17; these control measures will be further developed.

The following aspects of the generic SMR-300 radiological protection strategy, and specific SSCs, are examples which have been identified as having implications/relevance with regard to demonstration of both ALARP and BAT concepts.

- Minimisation of generation of radionuclides by design, including through items such as the heavy neutron reflector and material selection to reduce activation products and deposits in the RCS and those systems that connect to the RCS.
- SFP being located inside the containment structure reduces worker doses with inherent shielding properties.
- Water in the SFP is maintained low activity by means of a purification system.
- Use of high reliability fuel.
- CVC designed to prevent the accumulation of radionuclides in the primary coolant.
- Degassing of coolant water on start-up to minimise Ar-41 production (not so significant in terms of BAT due to relatively short half-life).

A fully integrated approach to the requirements associated with both BAT and ALARP will be undertaken, led by discipline leads and specialists, to demonstrate the application of both concepts to the generic SMR-300 design.

10.8.2.4 GDA Commitments and Forward Actions

There are no GDA commitments identified for PSR Chapter B10, Radiological Protection.

Forward actions have been collated and are managed via the process described in PSR Chapter A4 'Lifecycle Management of Safety and Quality Assurance' [19]. PSR Chapter A5 'ALARP Summary' [26] describes the contribution of the forward actions to the overall ALARP argument.

10.8.3 Conclusion

The conclusion of this PSR chapter is that:

- The chapter claims have been met to a maturity expected of a PSR. Further CAE can be provided as the design matures during and beyond the GDA.
- Safety functions have been identified for radiological protection SSCs in accordance with the SMR-300 design at this preliminary stage.

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- The application of the ALARA principle is broadly consistent with the application of ALARP in design, with some small differences relating to the limits and targets.
- There are further assessments that can be done during the GDA to provide qualitative figures to compare against numerical targets and hence, improve the demonstration of ALARP for the generic SMR-300.

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[Not Part 810 Export Controlled]

10.9 REFERENCES

[1] “Holtec Britain, "HI-2240332, Holtec SMR GDA PSR Part A Chapter 1 Introduction," Revision 0, August 2024”.

[2] “Holtec Britain, "HI-2240334, Holtec SMR GDA PSR Part A Chapter 3 Claims, Arguments and Evidence," Revision 0, August 2024”.

[3] “Holtec Britain, "HI-2240337, Holtec SMR GDA PSR Part B Chapter 1 Reactor Coolant System and Engineered Safety Features" Revision 0, August 2024”.

[4] Holtec Britain, “Holtec Britain, "HI-2240362, Holtec SMR GDA PER Chapter 3 Radiological Impact Assessment," Revision 0, August 2024”.

[5] “Holtec Britain, "HI-2240363, Holtec SMR GDA PER Chapter 4 Conventional Impact Assessment," Revision 0, August 2024”.

[6] “Holtec Britain, "HI-2240345, Holtec SMR GDA PSR Part B Chapter 14 Design Basis Accident Analysis," Revision 0, August 2024”.

[7] “Holtec Britain, "HI-2240355, Holtec SMR GDA PSR Part B Chapter 26 Decommissioning Approach," Revision 0, August 2024”.

[8] “Holtec Britain, "HI-2240333, Holtec SMR GDA PSR Part A Chapter 2 General Design Aspects and Site Characteristics," Revision 0, August 2024”.

[9] Holtec Britain, “SMR-300 Preliminary Safety Report for Generic Design Assesment, Chapter B2 - Reactor, Revision 0,” July 2024.

[10] “Holtec Britain, "HI-2240777, Holtec SMR GDA PSR Part B Chapter 5 Reactor Supporting Facilities," Revision 0, August 2024”.

[11] “Holtec Britain, "HI-2240338, Holtec SMR GDA PSR Part B Chapter 4 Control and Instrumentation Systems," Revision 0, August 2024”.

[12] “Holtec Britain, "HI-2240342, Holtec SMR GDA PSR Part B Chapter 11 Environmental Protection," Revision 0, August 2024”.

[13] “Holtec Britain, "HI-2240344, Holtec SMR GDA PSR Part B Chapter 13 Radioactive Waste Management," Revision 0, August 2024”.

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[Not Part 810 Export Controlled]

[14] “Holtec Britain, "HI-2240357, Holtec SMR GDA PSR Part B Chapter 20 Civil Engineering," Revision 0, August 2024”.

[15] “Holtec Britain, "HI-2240350, Holtec SMR GDA PSR Part B Chapter 21 External Hazards," Revision 0, August 2024”.

[16] “Holtec Britain, "HI-2240351, Holtec SMR GDA PSR Part B Chapter 22 Internal Hazards," Revision 0, August 2024”.

[17] “Holtec Britain, "HI-2240346, Holtec SMR GDA PSR Part B Chapter 15 BDBA, Severe Accidents Analysis and Emergency Preparedness," Revision 0, August 2024”.

[18] “Holtec Britain, "HI-2240347, Holtec SMR GDA PSR Part B Chapter 16 Probabilistic Safety Assessment," Revision 0, August 2024”.

[19] “Holtec Britain, "HI-2240335, Holtec SMR GDA PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance," Revision 0, August 2024”.

[20] “Holtec Britain, "HI-2240340, Holtec SMR GDA PSR Part B Chapter 9 Conduct of Operations," Revision 0, August 2024”.

[21] “Holtec Britain, "HI-2240354, Holtec SMR GDA PSR Part B Chapter 25 Construction and Commissioning," Revision 0, August 2024”.

[22] “Holtec Britain, "HI-2240343, Holtec SMR GDA PSR Part B Chapter 12 Control of Non-Radiological Hazards," Revision 0, August 2024”.

[23] “Holtec Britain, "HI-2240348, Holtec SMR GDA PSR Part B Chapter 17 Human Factors," Revision 0, August 2024”.

[24] “Holtec Britain, "HI-2240352, Holtec SMR GDA PSR Part B Chapter 23 Reactor Chemistry," Revision 0, August 2024”.

[25] “Holtec Britain, "HI-2240353, Holtec SMR GDA PSR Part B Chapter 24 Fuel Transport and Storage," Revision 0, August 2024”.

[26] “Holtec Britain, "HI-2240336, Holtec SMR GDA PSR Part A Chapter 5 Summary of ALARP," Revision 0, August 2024”.

[27] “HI-2240166-R0 - SMR-300 System Design Description for the Chemical and Volume Control System,” Holtec Britain, 2024.

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[Not UK Export Controlled]
[Not Part 810 Export Controlled]

[28] “HI-2230819-R0 - Primary Sampling System Requirements, Open Items, and Interface Requirements Notebook,” Holtec Britain.

[29] “HI-2146060-R0 - System Design Description for Main Control Room Habitability System,” Holtec Britain.

[30] “HI-2145972-R1 - System Design Description of Containment Cooling and Ventilation System,” Holtec Britain.

[31] “HI-2145999-R0 - Position Paper on HVAC Systems,” Holtec Britain.

[32] “HI-2200451-R1 - System Design Desciption of Gaseous Radwaste System,” Holtec Britain.

[33] “HI-2200449-R0 - System Design Description for Liquid Radwaste System,” Holtec Britain.

[34] “HI-2200695 – System Design Description for Solid Radwaste System,” Holtec Britain.

[35] “HI-2200826 – Radiation Monitoring System Detector Specification,” Holtec Britain.

[36] “HI-2177761-R2 - Design Specification for Containment Structure,” Holtec Britain.

[37] “HI-2188496-R1 - Design Specification for Containment Enclosure Structure,” Holtec Britain.

[38] “HI-2188323-R2 - Design Specification for Reactor Auxiliary Building,” Holtec Britain.

[39] Holtec SMR LLC, “HPP-160-3021, SMR-160 Design Standard for Radiation Protection”.

[40] Holtec International , “SMR-160 Preliminary Safety Analysis Report, Chapter 12 'Radiation Protection',” June 2023.

[41] International Commission on Radiological Protection, Annals of the ICRP Volume 21 No. 1-3, Publication 60, “1990 Recommendations of the International Commission on Radiological Protection”,, 1990.

[42] International Commission on Radiological Protection, “Annals of the ICRP Volume 37 Nos 2-4, IRP Publication 103, "2007 Recommendations of the International Commission on Radiological Protection",” 2007.

[43] International Atomic Energy Agency, “Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, General Safety Requirements Part 3,” Vienna, 2014.

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[Not UK Export Controlled]
[Not Part 810 Export Controlled]

[44] European Commission, Directorate-General for Energy, Basic Safety Standards Directive - Better Radiation Protection,, Publications Office, 2018.

[45] HM Government, The Environmental Permitting (England and Wales) Regulations 2016 SI 2016/1154, December 2016.

[46] Environmental Authorisations (Scotland) Regulations 2018, SSI 2018 No. 219, 2018.

[47] HM Government, The Ionising Radiations Regulations 2017 SI 2017/1075, January 2018.

[48] HM Government, The Radiation (Emergency Preparedness and Public Information) Regulations 2019, SI 2019 No. 703, May 2019.

[49] HM Government, The Justification of Practices Involving Ionising Radiation Regulations 2004, SI2004/1769, July 2004.

[50] US Federal Government, US Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation".

[51] US NRC, “NUREG-1736 Consolidated Guidance: 10 CFR Part 20 — Standards for Protection Against Radiation,” 2001.

[52] US NRC, “Regulatory Guide 1.8, Qualification and Training of Personnel for Nuclear Power Plants, Rev 4,” 2021.

[53] US NRC, “Regulatory Guide 8.8 - Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable, Revision 4,” 2021.

[54] US NRC, “Regulatory Guide 8.10 - Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable, Revision 3,” 1978.

[55] US NRC, “Regulation 10 CFR Part 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES”.

[56] US NRC, “10 CFR PART 100—REACTOR SITE CRITERIA,” 2015.

[57] US NRC, “StandardReview Plan for the Review of Safety Analysis Reports for Nuclear Power Plants:LWR Edition — Radiation Protection (NUREG-0800, Chapter 12),” 2013.

[58] US NRC, “Regulatory Guide 1.21 Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste, Rev 3,” Sept 2021. [Online].

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[Not UK Export Controlled]
[Not Part 810 Export Controlled]

[59] US NRC, “Regulatory Guide 1.45: Guidance on monitoring and responding to reactor coolant system leakage, Rev 1,” 2008.

[60] US NRC, “Regulatory Guide 1.69 - Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants, Revision 1,” 2009.

[61] US NRC, “Regulatory Guide 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,” 1977.

[62] US NRC, “Regulatory Guide 1.194 Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants , Rev 0,” 2003.

[63] US NRC, “Regulatory Guidance 1.196 Control Room Habitability at Light-Water Nuclear Power Reactors, Rev 1,” 2007.

[64] US NRC, “Regulatory Guide 1.197 Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors , Rev 0,” 2003.

[65] HSE, Work with ionising radiation, Ionising Radiations Regulations 2017, Approved Code of Practice and guidance, L121, 2018.

[66] HM Government, Nuclear Installations Act 1965, updated, 2023.

[67] HM Government, Health and Safety at Work etc. Act 1974, July 1974.

[68] HM Government, The Management of Health and Safety at Work Regulations 1999, SI 1999 No. 3242, 1999.

[69] HM Government, The Radiation (Emergency Preparedness and Public Information) Regulations 2019 Approved Code of Practice and guidance, L126, Nov 2020.

[70] Office for Nuclear Regulation, “Safety Assessment Principles for Nuclear Facilities, 2014 Edition Revision 1,,” 2020.

[71] Office for Nuclear Regulation, “NS-TAST-GD-038 – Radiological Protection, Issue 10,” Dec 2022.

[72] Office for Nuclear Regulation, “NS-TAST-GD-002 – Radiation Shielding, Issue 8.1”.

[73] Office for Nuclear Regulation, “NS-TAST-GD-004. Fundamental Principles, Issue 8”.

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[Not Part 810 Export Controlled]

[74] Office for Nuclear Regulation, “NS-TAST-GD-005 – Guidance on the Demonstration of ALARP, Issue 11.3”.

[75] Office for Nuclear Regulation, “NS-TAST-GD-022 – Ventilation, Issue 8.1”.

[76] Office for Nuclear Regulation, “NS-TAST-GD-024 – Management of Radioactive Material and Radioactive Waste on Nuclear Licensed Sites, Issue 7.2”.

[77] Office for Nuclear Regulation, “NS-TAST-GD-027 – Training and assuring personnel competence, Issue 7”.

[78] Office for Nuclear Regulation, “NS-TAST-GD-041 – Criticality Safety, Issue 8”.

[79] Office for Nuclear Regulation, “NS-TAST-GD-043 – Radiological Analysis – Normal Operation, Issue 6”.

[80] Office for Nuclear Regulation, “NS-TAST-GD-045 – Radiological Analysis – Fault Conditions, Issue 7”.

[81] Department of Energy and Climate Change, “The Energy Act 2008: Funded Decommissioning Programme Guidance for New Nuclear Power Stations”.

[82] Office for Nuclear Regulation, “NS-TAST-GD-100 – Shielding and Dose Rate Safety Assessment of Transport Packages, Issue 2”.

[83] Office for Nuclear Regulation, “NS-INSP-GD-054 - The Ionising Radiations Regulations 2017, Issue 8,” 2022.

[84] “ESOREX,” [Online]. Available: https://www.esorex-platform.org/.

[85] [Online]. Available: https://www.wano.info/resources/performance-indicators. [Accessed 28 Feb 2024].

[86] “ISOE Network - ISOE Annual Reports,” [Online]. Available: https://isoe-network.net/docs/publications/isoe-resources/isoe-annual-reports/. [Accessed 28 02 2024].

[87] [Online]. Available: https://docs.siseri.irsn.fr/rapports-dexpertises. [Accessed 28 02 2024].

[88] US NRC, “REIRS: Annual Report on Occupational Radiation Exposure, NUREG-0713,” [Online]. Available: https://www.reirs.com/0713.html. [Accessed 27 06 2024].

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[89] US NRC, “Human Factors Engineering Program Review Model, NUREG-0711, Revision 3,” 2012.

[90] Holtec Britain, “SMR-300 Preliminary Safety Report for Generic Design Assessment, Chapter B17 - Human Factors, Revision 0,” May 2024.

[91] IAEA, “SSG-90 Radiation Protection Aspects for Nuclear Power Plants,,” May 2024.

[92] Office for Nuclear Regulation, “Assessment of Reactors - Generic Design Assessment of New Reactors,” [Online]. Available: https://www.onr.org.uk/new-reactors/assessment.htm. [Accessed 04 03 2024].

[93] Holtec Britain, “HPP-3295-0017, Holtec SMR-300 Generic Design Assessment Reference Design Process and GDA Prospective Design Change Register, Revision 1,” 2024.

[94] Safety Directors Forum, “The Application of ALARP to Radiological Risk, A Nuclear Industry Good Practice Guide, Issue 1,” December 2012.

[95] US Nuclear Regulatory Commission, “Regulatory Guide 1.26 Revision 6 Quality Group Classifications and Standards For Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” 2021.

[96] Holtec Britain, “SMR-300 Preliminary Safety Report for Generic Design Assessment, Chapter B13 - Radioactive Waste Management, Revision 0,” May 2024.

[97] Holtec SMR LLC, “HI-22000637, SMR-160 Core Isotopics and Source Terms, Revision 2,” August 2023.

[98] Holtec International, “HI-2220151, SMR-160 Contained Radiation Sourcs for Normal Operations, Revision 4.0,” November 2023.

[99] Holtec Britain, “SMR-300 Preliminary Safety Report for Generic Design Assessment, Chapter B11 - Environmental Protection, Revision 0,” May 2024.

[100] Holtec Britain, “Approach and Application of BAT Demonstration , Revision 0,” May 2024.

[101] NuScale, “NuScale Standard Plant Design Certification Application, Chapter 12 Radiation Protection, Part 2-Tier 2, Revision 4.1,” 2020.

[102] International Organization for Standardization , “ISO 17873:2004 Nuclear Facilities - Criteria for the Design and Operation of Ventilation Systems for Nuclear Installations other than Nuclear Reactors,” 2004.

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[103] International Organization for Standardization, “ISO 26802:2010 Nuclear Facilities - Criteria for the Design and Operation of Containment and Ventilation Systems for Nuclear Reactors,” 2010.

[104] IAEA, “IAEA Nuclear Energy Series No. NW-T-1.38, Decontamination Methodologies and Approaches,” 2023.

[105] International Atomic Energy Agency, “IAEA-TECDOC-1657 Design Lessons Drawn from the Decommissioning of Nuclear Facilities,” 2011.

[106] “Changeroom Design, Operation and Maintenance, A Nuclear Industry Code of Practice, Issue 1,,” July 2006.

[107] Sellafield Limited, “ Ventilation systems for radiological facilities, design guide, ES_0_1738_1, Issue 2,” 2023.

[108] “ISO 17873:2004 ‘Nuclear facilities - Criteria for the design and operation of ventilation systems for nuclear installations other than nuclear reactors’,” 2004.

[109] “ISO 26802:2010 ‘Nuclear facilities - Criteria for the design and operation of containment and ventilation systems for nuclear reactors’,” 2010.

[110] Holtec International, “ HI-2240619 R0.0 System Design Description for Radiation Monitoring System,” 2024.

[111] Holtec International, “HI-2200826 SMR-160 Radiation Monitoring System Detector Specification, Revision 0,” 2021.

[112] Holtec International, “HI-2188373 SMR-160 Radiation Monitoring System Design Requirements Document, Revision 0,” 2019.

[113] IAEA, “Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition, Proceedings of a Technical Meeting held in London 29 August – 2 September 2005, IAEA-TECDOC-1547,” 2007.

[114] Nuclear Energy Agency, “Burn Up Credit and Nuclear Criticality Safety,” [Online]. Available: https://www.oecd-nea.org/jcms/pl_24448/burn-up-credit-and-nuclear-criticality-safety. [Accessed 04 03 2024].

[115] US NRC, “Regulatory Guide 8.19: Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage ManRem Estimates, Revision 1,” 1979.

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[Not UK Export Controlled]
[Not Part 810 Export Controlled]

[116] US NRC, “Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” February 1993.

[117] “Holtec Britain, "HI-2240341, Holtec SMR GDA PSR Part B Chapter 10 Radiological Protection," Revision 0, August 2024”.

[118] ONR, “Assessment of Reactors,” [Online]. Available: https://www.onr.org.uk/generic-design-assessment/assessment-of-reactors/. [Accessed 02 07 2024].

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10.10 LIST OF APPENDICES

Appendix A Relevant ROs A-1

Appendix B Relevant RIs B-1

Appendix C CAE Route Map C-1

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Appendix A Relevant ROs

All ROs for all reactor designs undertaking GDA can be found in ONR GDA Assessment of Reactors webpage [118] [92]. ROs relevant to Radiological Protection are presented below.

Table 15: Previous Regulatory Observations relevant to Radiological Protection

Regulatory Observation	Notes on RO	Notes on Resolution Plan	Lessons Learned
UK HPR 1000			
RO–UKHPR1000-28 Adequate Justification of Estimated Public Doses for UK HPR1000	Dose needs to be estimated accurately to support RP's claim that doses to members of public from direct radiation shine have been reduced SFAIRP. RP needs to demonstrate that public doses from a comparable reactor can be compared to UK HPR1000 and analyse/explain any differences. RP should consider impact of other facilities e.g., radioactive waste and spent nuclear fuel stores.	Document to demonstrate doses to Members or the Public are no greater than background based on historical site information. Updated Public Dose Evaluation from Direct Radiation Topic Report, drawing conclusions for UK HPR1000, considering impact from ILW and spent fuel Storage Facilities; demonstrates that environmental results support argument that doses to members of public from direct radiation are reduced SFAIRP.	OPEX from existing ISFSI sites gathered to support argument that there will be no detectable radiation dose (above background) at the site boundary. Review to be undertaken to identify candidate comparable reactors to compare exposures.
RO–UKHPR1000-35 Optimisation of collective occupational radiation exposure for the UK HPR1000	Dose data presented for UK HPR1000 did not compare favourably with PWRs worldwide, therefore, RP needed to demonstrate that collective occupational radiation exposure had been suitably optimised and reduced to ALARP. Consider ONR SAPs: RP.1 & 7. ONR expected a robust demonstration that UK HPR1000 design and intended operations have both been optimised such that predicted doses are comparable to or lower than current average for PWRs worldwide.	Gap analysis to assess differences between Chinese PWRs and UK regarding occupational exposure and to explain how occupational exposure on UK HPR1000 is expected to be influenced. Provide evidence by ALARP documents produced by SSC design areas and Radiological Protection areas to demonstrate application of ALARP process for Radiological Protection to minimise/reduce occupational exposure to ALARP. New documents: Establishment of the Starting Point for the Collective Dose Evaluation of the UK HPR1000; EIMT Consistency Analysis; Evaluation of the Impacts on Collective Dose from the Design Improvements.	Gap analysis has been carried out to compare occupational doses from US PWR reactors to UK and European reactors to identify differences in dose trends. Further detailed comparison planned as part of task analysis. Identification of these differences will enable design and operational improvements to be incorporated to further reduce exposures SFAIRP.

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Regulatory Observation	Notes on RO	Notes on Resolution Plan	Lessons Learned
RO–UKHPR1000-60 Scope and Plan for Radiation Shielding Assessments for the UK HPR1000 Design	Gaps identified: Ambiguity regarding RP's planned work for GDA to assess penetrations through bulk radiation shielding for UK HPR1000 generic design; Shielding design reports did not clearly state the dose rates presented omit contributions from penetrations; Insufficient evidence to demonstrate method for identifying and sealing penetrations was adequate; No evidence, examples or OPEX to demonstrate adequacy of penetration shielding design. Relevant Legislation, Standards and Guidance: IRR17 Regulation 9 paragraphs 101, 102, 104, 107 & 108; ONR SAPs RP.6 & 7; TAG-002 Radiation Shielding sections 4.7, 4.8, 5.30 & 5.31. ONR expectations: 1) Clear identification and justification of shielding assessments planned/already completed during and post-GDA for UK HPR1000; 2) Clear documentation of method(s) for identifying/assessing radiation shielding penetrations in UK HPR1000 generic design and justification why they show good practice and fit-for-purpose.	New document: Radiation Shielding Assessment Strategy and Scope for the UK HPR1000 Generic Design to demonstrate, by provision of suitable references, that the shielding provisions for bulk shielding, openings shielding, local shielding and transient source term shielding have been completed for GDA. Updated reports: Penetrations Shielding Design Report to ensure all transmission paths have been adequately captured for GDA and demonstrate that potential changes in dose rates resulting from detailed penetration shielding design post-GDA won't impact radiation zoning and layout design; Radiological Protection Design Principles of Opening; UK HPR1000 shielding design reports & Radiation Shielding Topic Report to conclude that potential changes in dose rates resulting from detailed penetration shielding design post-GDA won't significantly impact radiation zoning and layout design.	At PSR stage, only bulk shielding will be assessed, however the design has already considered techniques for mitigating penetrations in shielding. High level radiation zoning will be carried out as part of the GDA to set the dose rate targets for plant areas that bulk shielding, and any penetrations will need to achieve.
UK ABWR			
RO-ABWR-14 UK ABWR Radiological Protection Safety Case: Project Plan and Delivery	Radiological Protection safety case was reactive rather than proactive. Main concerns: overall objectives, strategy and scope of the safety case not stated; uncertainties how integration of Radiological Protection aspects in overall UK ABWR safety were to be managed to ensure coherent and proportionate safety case demonstration; no full consideration of hierarchy of submissions constituting to Radiological Protection safety case and their structure; unclear how radiological protection aspects of safety case will developed by submissions identified; unclear how any Radiological Protection supporting documents produced will be used to support PCSR	Hitachi-GE were to provide their strategy for development of the Radiological Protections elements of the UK ABWR Safety Case. Strategy may need to be updated throughout GDA - any significant changes to strategy will be agreed with ONR before implementing. Hitachi-GE were to provide a plan for delivery of the Radiological Protections elements of the UK ABWR Safety Case. Hitachi-GE were to allocate SQEP to develop the Radiological Protection safety case.	Following review of this RO, further development of the scope radiological protection at the PSR stage was completed to ensure a proactive integration of Radiological Protection into the design and safety case. Radiation Protection advisers, with experience of safety case, nuclear design and operation of nuclear facilities lead on the radiological protection chapter.
RO-ABWR-25 Safety Case Process and Capability	Safety case needs to be complete, cogent and coherently developed, and take into account UK regulatory expectations. Inconsistencies were identified amongst various and technical safety case documentation, and its formulation into a coherent safety case that takes cognisance of the multiple relevant interfaces.	Action 1: Work closely with the Regulators to develop Safety Case to demonstrate relevant standards have been met and risks reduced ALARP. Produced revised Safety Case Development Plan and Safety Case Manual. Action 2: Define and secure suitably qualified and experienced safety case professionals. Establishment of an expert Safety Case Steering Group (SCSG). A letter to the Regulators confirming the SCSG is operational and summary of the Consultant team supporting preparation of the Safety Case. Action 3: PCSR Safety Case Manager needs to understand UK ABWR Technology. Role Profile for the Safety Case Supervisor and Terms of Reference (TOR) for the Safety Case Steering Group.	Interfaces with other chapters of the PSR and PER were identified early in the PSR process, multidiscipline reviews of the chapter by the authors for these interfacing topics to ensure consistence across the safety case. UK regulatory expectations are identified together with plans for meeting these expectations.

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Regulatory Observation	Notes on RO	Notes on Resolution Plan	Lessons Learned
RO-ABWR-45	Raised to ensure RP's safety case submission demonstrates adequate consideration to relevant operational experience. RP had not taken into account OPEX from BWRA plants from around the world including Japan. GDA needs to 1) baseline ONR's regulatory knowledge of all BWR and ABWR OPEX across the world; 2) demonstrate adequacy and robustness of ABWR technology; 3) show adequately how OPEX has been considered to reduce risks SFAIRP.	Initiate a review of OPEX of Japanese BWR nuclear power plants and combine this information with the 2 reports commissioned by ONR to form a summary report of OPEX (OPEX Report for UK ABWR) to be used to identify generic themes and trends of relevant OPEX. Prepare a summary/synopsis on how it will demonstrate UK ABWR design and safety case considers Japanese and relevant international OPEX. Submission of first OPEX study result. OPEX should be reviewed and revised regularly during and after GDA until the last UK ABWR will be completely decommissioned.	The SMR-300 is a novel PWR reactor design with no directly relevant OPEX from other similar SMRs, therefore a plan will be developed to identify the most relevant sources of OPEX from both civil nuclear and military reactor designs. Key concerns for relevance of OPEX are the compact nature of a SMR, scaling in reactor size to magnitude of dose and risk, and the extended planned operating lifetime of the reactor compared to those currently in operation or decommissioning.
RO-ABWR-64 Design approach to identification and provision of both permanent and temporary features necessary for the adequate control of radioactive contamination across the full lifetime of UKABWR	Hitachi-GE's approach to control radioactive contamination was unclear. UKABWR should be designed such that permanent and temporary features required to manage and prevent the spread of radioactive contamination, from areas of high designation to those of lower designation are fully considered. Requested Hitachi-GE to show how it will implement a design approach that meets ONR expectations for the design of UK ABWR related to contamination control.	Documents to identify: Location of expected potential contamination; - Events with possibility of contamination spread, e.g., opening radioactive equipment for maintenance & sampling, piper rupture during faults/accidents, and pipe disconnection for decommissioning; - Nature (S, L & G) & Extent (Volume & activity) of potential contamination. Contamination Control Philosophy Document. Contamination Control and Protection against Direct Radiation: Design Study Document(s). Strategy for access control to high dose rate and high dose areas. Access control to high dose rate and high dose areas (representative examples).	A review of the SSCs and design features incorporated to provide radioactive contamination control in the SMR-300 design will be carried out, comparing the design against identified RGP to highlight and differences required for the generic SMR-300 design to meet UK standards. Sources of contamination will be determined from literature review of OPEX from other nuclear facilities.
RO-ABWR-65 Demonstration of adequate design and implementation of inherently safe techniques and structures to minimise radiation dose rates via through wall penetrations during all operating modes and for the lifetime of the facility, whilst being cognisant of design requirements relating to other discipline areas.	Shortfalls identified during Step 3 related to proposed solution for penetrations where lead wool was to be used, and to perceived lack of integrated approach the design of penetrations which would demonstrate reduction of exposures SFAIRP. It is essential to consider other hazard potentials and potentially competing requirements from other specialisms in the design of penetrations through shield walls. All operating modes and for lifetime of facility needs considering too. It is vital to ensure penetrations are adequately designed to either remove potential for elevated dose rates in adjacent areas or minimise this to a level considered ALARP. Consider other disciplines/specialisms and their associated hazards. Design should be inherently safe. SAPs Require Good Engineering principles to be applied: EKP1-EKP3.	Scope of work: 1) Hitachi-GE to identify the geometry of the penetrations and level of relevant hazards; 2) provide ONR with a strategy for design such that shine paths through the penetrations are suitably attenuated to ensure dose rates are reduced SFAIRP; 3) provide evidence that the strategy ensures engineering designs are inherently safe and utilise the hierarchy of controls focusing on passive safety. Deliverables: Penetration Design Guideline Document, Penetration Design Study Document(s), The relevant PCSR(s), Topic Report(s) and Supporting Document(s) were to be updated, as necessary.	As above, at this design stage the number, size and positioning of any penetrations is unknown, therefore the impacts cannot be determined. Penetration assessments will be incorporated into the PCSR taking cognisance of all operating modes and the whole lifetime of the facility.

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Regulatory Observation	Notes on RO	Notes on Resolution Plan	Lessons Learned
RO-ABWR-73 Robust demonstration that the design of the UK ABWR off-gas system reduces risks SFAIRP	RO was raised jointly by ONR & EA as Off-gas system is a principal discharge route to atmosphere during normal operations, where requirement to apply BAT to discharges and disposals of radioactive waste applies. There were a number of gaps between Hitachi-GE's submissions and regulatory expectations of off-gas system. The way in which Hitachi-GE's 'non-reactor' categorisation and classification scheme for Structures, Systems and Components (SSCs) had been applied to the off-gas system could be challenged. Thought should be given to the robustness of conclusions of ALARP demonstration with respect to uncertainties and to any assumptions employed in the demonstration. Where cases use quantitative methods, sensitivity studies to test robustness of the arguments need providing. Since off-gas system was located outside of main containment vessel, it represented a potential source of increased radioactivity available for release to the environment should the system fail to perform as intended or due to fault scenarios. Therefore, demonstration of RGP to reduce all relevant risks to ALARP was necessary. NS-TAST-GD-005 provide ONR's expectations with respect to demonstrating ALARP. HSE have published a guidance on ALARP. ONR expect inclusion of the following: Application of RGP, Options and optioneering (record range of options included and discarded to aid transparency in ALARP demonstration), Known problem areas, Proper balancing of risks, Taking cognisance of all relevant legislation & Uncertainties and the precautionary principle.	Hitachi-GE to implement a gap analysis based on options derived from RGP & worldwide OPEX to eliminate, reduce or mitigate risks associated with reference design SFAIRP, whilst ensuring appropriate use of BAT. Topic Report on ALARP Assessment for Off-Gas System were to be updated through: Development of the "route map" for ALARP demonstration, Understanding of RGP and worldwide OPEX, Review of existing risk/hazard assessment results, Alignment with relevant UK legal requirements, Gap analysis of baseline Off-Gas system design against RGP, Revised ALARP demonstration. Include more information about good practice and OPEX in the updated ALARP report. ALARP report was to be updated to include justification of the containment function and reduction of radioactive wastes.	The off-gas system (GRW) is under development for the SMR-300, HSE and ONR guidance on ALARP have been reviewed in support of the PSR. A method [93] has been developed to ensure that BAT and ALARP are appropriately incorporated into design decisions for the generic SMR-300.

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Regulatory Observation	Notes on RO	Notes on Resolution Plan	Lessons Learned
RO-ABWR-75 Robust demonstration that the design of the UK ABWR HVAC system has been adequately conceived and reduces risks SFAIRP	Observations & expectations: SSCs, should be categorised and classified on the basis of their safety significance. Interfaces between SSC of different category and class should be designed to ensure that any failure in a lower category and class item will not adversely impact an SSC of a higher category and class. No evidence that segregation had been considered by the responsible designer to ensure failure independence. The design basis for environmental limits and conditions for each building was not substantiated by arguments or evidence. No justification of how HVAC protects personnel and equipment against specific risks from inside/outside buildings. BoSC did not provide a robust justification for the classification of the humidifier units nor an impact assessment for loss of humidifier functionality on nuclear safety systems. Very specific performance figures were quoted with no link to evidence to substantiate the design basis. HVAC safety documentation did not consider the requirement for HVAC systems to support POCO or decommissioning phases of the facility lifecycle (ONR SAP DC.1 Decommissioning should be taken into account during the planning and design of a new facility). Qualification provisions did not align with regulatory expectations (ONRs SAP EQU (Equipment Qualification) and TAG NS-TAST-GD-094). Configuration of safe change filter units forming a “U Shape” should not be optimal configuration - potential radiation exposure from almost a full 360 degrees. Opportunity to provide local extract ventilation wasn't explored in HVAC design.	Revised Basis of Safety Case on HVAC.	The safety related SSCs for HVAC are identified and have undergone categorisation and classification under the US scheme. At this design stage, the layout of vent plant has not been defined. Radiation Protection and Mechanical Engineering disciplines work closely together to assess the HVAC design against UK RGP.

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Appendix B Relevant RIs

All RIs for all reactor designs undertaking GDA can be found in the following reference [92]. RIs relevant to Radiological Protection are presented below.

Table 16: Previous Regulatory Issues relevant to Radiological Protection

Regulatory Issue	Notes on RI	Resolution Plan Notes	Lessons Learned
UKABWR			
RI-ABWR-01 Definition and Justification for the Radioactive Source Terms in UK ABWR during Normal Operations	Assumptions made in calculation of source terms were not appropriately justified. Scope of defined source terms incomplete. No link between source terms and extant UK ABWR safety & environmental cases. A suitably robust demonstration and justification for the adequacy of the defined source terms had not been provided. Definition of source term should cover/include: -all significant radionuclides; -all systems which are expected to contain radioactivity; -all operational states; -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; -all aspects of the safety or environmental case for UK ABWR; -be consistent with how the defined source terms are used by, and support, these cases; -be consistent with the design and operations of UK ABWR. Justification of source term should: -provide an appropriate degree of robust supporting evidence for the defined source terms; cover the full scope of the definition, but be targeted towards those radionuclides, systems or operations which have the highest safety or environmental impact; -be demonstrated to be appropriate for the UK ABWR and consistent with the extant safety and environmental cases.	New document structure to provide a clearer understanding of how the Source Term will be used and effectively integrated into the UK ABWR safety and environmental cases: 1) Strategy document; 2) Source term manual; 3) Source term values; 4) Supporting source term manual references.	As source term interfaces with many disciplines, it is essential that it integrates throughout both the PER and PSR. An integrated source term document is undergoing production. This will provide the methods for determining each of the source terms to be incorporated into each relevant chapter together with the assumptions, input parameters and developed source term. The document will highlight the key radionuclides that will feed into individual assessments. The source term document will have multiple iterations as the design develops and assessments show where further design improvements can be made to minimise the source term.

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Appendix C CAE Route Map

Table 17: Chapter B10 CAE Route Map

Overarching SSEC Claim:	Overarching SSEC Sub Claim:	Chapter Claim:	Chapter Sub Claim:	Subchapter
Claim 2.1 Nuclear Safety 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 As Low As Reasonably Practicable.		**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 2.1.1.1** Effective dose targets, constraints and limits are identified for normal operations.	10.5 Radiological Protection Requirements
			Claim 2.1.1.2 Effective doses to workers and members of the public during normal operations are assessed.	10.7 Dose Assessments for Workers and members of the Public
			Claim 2.1.1.3 Effective doses to workers and members of the public during normal operations are demonstrated to be tolerable and ALARP.	10.6 Design for Radiological Protection
Claim 2.2 System/Process Design and Substantiation The design of the systems and associated processes are developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.	**Claim 2.2.3 Critical Safety Function 3** Adequate provision for the control of radiation exposure and control of release of radioactive material is incorporated into the design.	**Claim 2.2.3.4** SSCs are designed to meet Radiological Protection requirements and minimise exposures.	**Claim 2.2.3.4.1** Criticality controls are implemented to ensure that criticality risks are reduced to As Low As Reasonably Practicable.	10.6.9 Criticality Controls
			Claim 2.2.3.4.2 The Generic Holtec SMR-300 shielding is designed and substantiated to minimise exposures for all plant areas and operation stages, including waste package transport.	10.6.3 Shielding Design
			Claim 2.2.3.4.3 Materials selection minimises the generation of neutron activation products in SSCs	10.6.5 Material Selection
			Claim 2.2.3.4.4 Ventilation and containment systems are designed to meet Radiological Protection requirements and minimise exposures.	10.6.7 Ventilation design
			Claim 2.2.3.4.5 The design and layout of the Generic Holtec SMR-300 includes monitoring and alarm systems to alert to the presence of radiation and contamination.	10.6.6 Layout design

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Overarching SSEC Claim:	Overarching SSEC Sub Claim:	Chapter Claim:	Chapter Sub Claim:	Subchapter
			Claim 2.2.3.4.6 Radiation zoning ensures access to and between Controlled and Supervised Areas will be suitably controlled to minimise exposures to As Low As Reasonably Practicable and prevent the spread of radioactive material.	10.6.4 Area Designation, Radiation and Contamination Zoning

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