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Pre-Construction Safety Report

Chapter 22

Radiological Protection

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TABLE OF CONTENTS

22.1 List of Abbreviations and Acronyms

ACoP	Approved Code of Practice (UK)
ALARP	As Low As Reasonably Practicable
APG	Steam Generator Blowdown System [SGBS]
BAT	Best Available Technique
BFX	Fuel Building
BNX	Nuclear Auxiliary Building
BPX	Personnel Access Building
BQF	Spent Fuel Interim Storage Facilities
BQS	Waste Auxiliary Building
BQZ	Interim Storage Facility for Intermediate Level Waste
BRX	Reactor Building
BSA	Safeguard Building A
BSB	Safeguard Building B
BSC	Safeguard Building C
BSL	Basic Safety Level
BSO	Basic Safety Objective
BSS	Basic Safety Standard
BWX	Radioactive Waste Treatment Building
CGN	China General Nuclear Power Corporation
CPR1000	Chinese Pressurised Reactor
CRDM	Control Rod Drive Mechanism
CVI	Condenser Vacuum System [CVS]
DBA	Design Basis Accident
DBC	Design Basis Condition
DEC-A	Design Extension Condition A
DEI	Dose Equivalent Iodine
EBA	Containment Sweeping and Blowndown Ventilation System

	[CSBVS]
ECS	Extra Cooling System [ECS]
ERIC/PPE	Eliminate, Reduce, Isolate, Control, Personal Protective Equipment
EUF	Containment Filtration and Exhaust System [CFES]
GDA	Generic Design Assessment
HPR1000 (FCG3)	Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3
HSE	Health and Safety Executive (UK)
HSW Act	Health and Safety at Work etc Act 1974 (UK)
HVAC	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEC	International Electrotechnical Commission
IRR17	Ionising Radiations Regulations 2017
IRWST	In-containment Refuelling Water Storage Tank
ISOE	Information System on Occupational Exposure
KRT	Plant Radiation Monitoring System [PRMS]
KZC	Controlled Area Access Monitoring System [CAAMS]
MCR	Main Control Room
OECD	Organization for Economic Co-operation and Development
ONR	Office for Nuclear Regulation (UK)
OPEX	Operating Experience
PCER	Pre-Construction Environmental Report
PCSR	Pre-Construction Safety Report
PMC	Fuel Handling and Storage System [FHSS]
PPE	Personal Protective Equipment
PSR	Preliminary Safety Report
PTR	Fuel Pool Cooling and Treatment System [FPCTS]

PWR	Pressurised Water Reactor
RCP	Reactor Coolant System [RCS]
RCV	Chemical and Volume Control System [CVCS]
REA	Reactor Boron and Water Makeup System [RBWMS]
REN	Nuclear Sampling System [NSS]
REPPIR	Radiation (Emergency Preparedness and Public Information) Regulations
RGP	Relevant Good Practice
RIS	Safety Injection System [SIS]
RPE	Nuclear Island Vent and Drain System [VDS]
RRI	Component Cooling Water System [CCWS]
SA	Severe Accident
SAP	Safety Assessment Principle (UK)
SEL	Conventional Island Liquid Waste Discharge System [LWDS (CI)]
SSC	Structures, Systems and Components
TEG	Gaseous Waste Treatment System [GWTS]
TEP	Coolant Storage and Treatment System [CSTS]
TER	Nuclear Island Liquid Waste Discharge System [NLWDS]
TES	Solid Waste Treatment System [SWTS]
TEU	Liquid Waste Treatment System [LWTS]
TR	Topic Report
UK HPR1000	UK version of the Hua-long Pressurised Reactor
VVP	Main Steam System [MSS]
WENRA	Western European Nuclear Regulators Association

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Steam Generator Blowdown System (APG [SGBS]).

22.2 Introduction

This chapter of the Pre-Construction Safety Report (PCSR) summarises the radiological protection safety case for UK version of the Hua-long Pressurised Reactor (UK HPR1000). It covers normal plant operation, i.e. start-up, power operation, shutdown, outage, maintenance and inspection. It also describes the measures in place to reduce exposure of workers to radiation and contamination during accident mitigation.

This chapter identifies the radiological hazards associated with normal operation of the UK HPR1000 and defines the radioactive sources during normal operation that have been considered. It also defines the strategy to ensure that the exposure to radiation is As Low As Reasonably Practicable (ALARP), and the protection measures that have been considered in the design to protect against direct radiation and radioactive contamination. This chapter also includes description of radiation and contamination monitoring for occupational exposure.

This chapter demonstrates that the radiation dose to workers (for both internal and external doses) and to members of the public (from direct radiation) during normal operation complies with UK legal requirements and is ALARP.

PCSR Chapter 22 is developed based on the Preliminary Safety Report (PSR) Chapter 22, which focuses on the understanding of UK context and general considerations made to radiological protection design for the Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)).

22.2.1 Chapter Route Map

The ***Fundamental Objective*** of the UK HPR1000 is that: *The Generic UK HPR1000 could be constructed, operated, and decommissioned in the UK on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the environment.*

To underpin this objective**,** five high level claims and a number of level 2 claims are developed and presented in Chapter 1. Chapter 22 supports the ***Claim 3.4*** which is derived from the high level ***Claim 3***.

Claim 3*: The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable.*

Claim 3.4*: The safety assessment shows that the nuclear safety risks are ALARP.*

Based on the above high level claim and level 2 claim, two top claims for radiological protection safety case are:

Claim 3.4.4*: The risk to workers and members of the public from the potential* *harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP.*

Claim 3.4.5*: The risk to workers and members of the public from the potential harmful effects of ionising radiation resulting from fault and accident conditions complies with UK legal requirements and is ALARP.*

The object of risk evaluation mentioned in these two claims includes the following groups:

a) Onsite workers;

b) Members of the public.

The arguments and evidence to demonstrate that the risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP is provided in PCSR Chapter 22.

To demonstrate that the risk to members of the public from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP, two main arguments, along with supporting evidence, are presented:

a) One of the main routes for controlling public exposure is minimising radiation levels from direct radiation, which is provided in Sub-chapter 22.10;

b) Another main route of controlling public exposure is minimising the discharge of radioactive material, which is provided in *Pre-Construction Environmental Report (PCER) Chapter 7: Radiological Assessment*, Reference [1].

In order to support ***Claim 3.4.4***, this chapter develops two sub-claims and its corresponding arguments:

a) ***Sub-claim 3.4.4.1****: The risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP.*

   1) ***Argument 1.1****: The sub-claim 3.4.4.1 is demonstrated through consideration of measures to minimise radiation levels and reduce the risk to workers with regards to external exposure and internal exposure;*

   2) ***Argument 1.2****: The sub-claim 3.4.4.1 is demonstrated through consideration of measures to minimise exposure time and reduce the risk to workers with regards to external exposure and internal exposure;*

   3) ***Argument 1.3****: The sub-claim 3.4.4.1 is demonstrated through consideration of measures to minimise contamination levels and reduce the risk to workers with regards to external exposure and internal exposure.*

b) ***Sub-claim 3.4.4.2****: The risk to members of the public from direct radiation during normal operation complies with UK legal requirements and is ALARP.*

   1) ***Argument 2.1****: The sub-claim 3.4.4.2 is demonstrated through consideration of measures to minimise radiation levels and reduce the risk to members of the public from direct radiation.*

For worker dose evaluation under fault and accident conditions, workers can be categorised as follows: workers mitigating fault/accident conditions, and other workers present during fault/accident conditions. In this chapter, the dose assessments for workers mitigating fault/accident conditions are covered in the sub-chapter on post-accident accessibility. Dose assessments for other workers and members of the public under fault and accident conditions are presented in PCSR Chapter 12, Chapter 13 and Chapter 14.

In order to support ***Claim 3.4.5***, this chapter develops one sub-claim and its corresponding argument:

a) ***Sub-claim 3.4.5.1****: The risk to workers mitigating fault/accident conditions complies with UK legal requirements and is ALARP.*

   1) ***Argument 1.1****: The sub-claim 3.4.5.1 is demonstrated through consideration of all practicable measures to reduce exposure to radiation and contamination for workers who are mitigating fault and accident conditions (as well as dose assessment to such workers).*

The radiological protection safety case is described in Sub-chapter 22.4.

22.2.2 Chapter Structure

The structure of PCSR Chapter 22 consists with the following sub-chapters:

Sub-chapter 22.1 List of Abbreviations and Acronyms: This sub-chapter lists the abbreviations, acronyms used in the PCSR Chapter 22.

Sub-chapter 22.2 Introduction: This sub-chapter provides the overall layout of the chapter and brief descriptions of contents.

Sub-chapter 22.3 Applicable Codes and Standards: This sub-chapter provides a list of UK legislation, codes and standards relevant to radiological protection.

Sub-chapter 22.4 Regulatory Requirements and Safety Case: This sub-chapter demonstrates an understanding of UK regulatory requirements and defines the structure of the radiological protection safety case.

Sub-chapter 22.5 Source Term: This sub-chapter describes the definition of radioactive sources for UK HPR1000 and covers the various source terms for normal operation.

Sub-chapter 22.6 Strategy for Ensuring Exposure is ALARP: This sub-chapter provides a methodology for ensuring that radiation doses to workers and members of the public are ALARP.

Sub-chapter 22.7 Radiological Protection Measures: This sub-chapter summarises protection and provisions used against direct radiation and contamination control for the UK HPR1000.

Sub-chapter 22.8 Radiation and Contamination Monitoring: This sub-chapter presents plant radiation monitoring and radioactive contamination monitoring.

Sub-chapter 22.9 Dose Assessment for Workers: This sub-chapter presents the dose assessment for on-site workers based on the strategy for ensuring exposure is ALARP.

Sub-chapter 22.10 Dose Assessment for Public from Direct Radiation: This sub-chapter presents the evaluation of external dose to members of the public from direct radiation based on the strategy for ensuring exposure is ALARP. An overall dose assessment to members of the public during normal operation is covered by *PCER Chapter 7: Radiological Assessment*, Reference [1].

Sub-chapter 22.11 Post-accident Accessibility: This sub-chapter presents the post-accident accessibility requirements and the relevant dose assessment for workers participating in accident mitigation.

Sub-chapter 22.12 Concluding Remark: This sub-chapter presents the conclusion of PCSR Chapter 22.

Sub-chapter 22.13 References: This sub-chapter lists the references in PCSR Chapter 22.

22.2.3 Interfaces with Other Chapters

The interfaces with other PCSR chapters are listed in the following table.

T-22.2-1 Interfaces between Chapter 22 and Other Chapters

PCSR Chapter	Interface
Chapter 1 Introduction	Chapter 1 provides the fundamental objective, level 1 claims and level 2 claims. Chapter 22 provides chapter claims and arguments to support the relevant claims in Chapter 1.
Chapter 4 General Safety and Design Principles	Chapter 4 presents the radiation safety requirements. Chapter 22 presents the design of radiological

PCSR Chapter	Interface
	protection based on the radiation safety requirements set out in Chapter 4.
Chapter 5 Reactor Core	Chapter 5 provides reactor core design information used in source term design. Chapter 22 provides the definition of radioactive sources for the UK HPR1000 and covers the various source terms for normal operation.
Chapter 6 Reactor Coolant System	Chapter 6 provides reactor coolant system design information used in radiological protection design. Chapter 22 provides radiological protection design considerations relevant to the reactor coolant system substantiation.
Chapter 7 Safety Systems	Chapter 7 provides safety systems design information used in radiological protection design. Chapter 22 provides radiological protection design considerations relevant to the safety systems substantiation.
Chapter 10 Auxiliary Systems	Chapter 10 provides auxiliary systems design information used in radiological protection design. Chapter 22 provides radiological protection design considerations relevant to the nuclear auxiliary systems substantiation.
Chapter 11 Steam and Power Conversion System	Chapter 11 provides steam and power conversion system design information used in radiological protection design. Chapter 22 provides radiological protection design considerations relevant to the steam and power conversion system substantiation.
Chapter 12 Design Basis Condition Analysis	Chapter 12 provides design basis accident source terms. Chapter 22 provides the primary coolant source

PCSR Chapter	Interface
	term for radiological consequence analysis under Design Basis Condition (DBC) accidents.
Chapter 13 Design Extension Conditions and Severe Accident Analysis	Chapter 13 provides Design Extension Condition A (DEC-A) and severe accident source terms.
Chapter 17 Structural Integrity	Chapter 17 provides optimum material selection on minimisation of source term. Chapter 22 provides radiological protection design considerations relevant to material selection.
Chapter 20 MSQA and Safety Case Management	Chapter 20 provides the organisational arrangements and quality assurance arrangements which are implemented in the design process and in the production of this chapter.
Chapter 21 Reactor Chemistry	Chapter 21 provides optimum reactor water controls for the minimisation of source term. Chapter 22 provides the definition of radioactive sources for UK HPR1000 and covers the various source terms for normal operation.
Chapter 23 Radioactive Waste Management	Chapter 23 provides radioactive waste management systems design information used in radiological protection design. Chapter 22 provides source terms and the general radiological protection considerations for the radioactive waste management systems.
Chapter 24 Decommissioning	Chapter 24 provides source terms for the decommissioning assessment. Chapter 22 provides the definition of radioactive sources for the UK HPR1000 and covers the various source terms for normal operation.
Chapter 28 Fuel Route and Storage	Chapter 22 provides the dose assessment for workers and the general radiological protection

PCSR Chapter	Interface
	measures against direct radiation and radioactive contamination during fuel handling and storage related normal operation.
Chapter 29 Interim Storage of Spent Fuel	Chapter 22 provides the general radiological protection measures against direct radiation and radioactive contamination.
Chapter 31 Operational Management	Chapter 31 provides optimum operating procedures on the minimisation of source terms.
Chapter 32 Emergency Preparedness	Chapter 32 provides emergency preparedness including emergency operations, on-site accident management and emergency facilities. Chapter 22 presents information on post-accident accessibility which is connected with emergency operations.
Chapter 33 ALARP Evaluation	Chapter 33 consolidates the overall ALARP demonstration including the assessment of the radiation protection targets. Chapter 22 provides the assessment and ALARP demonstration of workers and members of the public from direct radiation during normal operation which supports the overall ALARP demonstration.

22.3 Applicable Codes and Standards

The codes and standards applied in radiological protection comply with the current existing requirements of applicable UK acts, regulations and other statutory instruments. According to the selection principles in PCSR Chapter 4 and *General Principles for Application of Laws, Regulations, Codes and Standards*, Reference [2], the main selection criteria for applicable codes and standards for radiological protection are as follows:

a) The Relevant Good Practice (RGP) of international organisations (e.g. International Atomic Energy Agency (IAEA), Western European Nuclear Regulators Association (WENRA) etc.) or other countries (e.g. UK, US, France etc.), as acknowledged by UK regulators, is taken into account sufficiently;

b) Based on experience to date, the selection of guidance documents, design codes and standards used in other Generic Design Assessment (GDA) projects are taken into account sufficiently;

c) In general, applicable relevant good practice, as recognised by UK regulators, is selected;

d) The latest version of guidance documents, design codes and standards are adopted.

Based on the above principles, the codes and standards adopted for radiological protection design include UK regulations and the Approved Code of Practice (UK) (ACoP), and international guidance documents, such as IAEA standards and International Commission on Radiological Protection (ICRP) publications.

The regulations for radiological protection in the UK are governed by the *Ionising Radiations Regulations 2017 (IRR17)*, Reference [3], and the *Approved Code of Practice*, Reference [4].

The IAEA safety standards and ICRP publications are examples of RGP that are recognised by Office for Nuclear Regulation (UK) (ONR).

International and UK codes and standards identified in T-22.3-1 have been considered in Chapter 22.

T-22.3-1 Applicable Codes and Standards for Radiological Protection

Standards Number	Title
---	The Health and Safety at Work etc. Act 1974
---	The Ionising Radiations Regulations 2017
---	Work with ionising radiation, Ionising Radiations Regulations 2017, Approved Code of Practice and guidance
---	The Radiation (Emergency Preparedness and Public Information) Regulations 2001
ICRP Publication 103	The 2007 Recommendations of the International Commission on Radiological Protection
GSR Part 3	Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards

Standards Number	Title
SSR-2/1	Safety of Nuclear Power Plants: Design
NS-G-1.13	Radiological Protection Aspects of Design for Nuclear Power Plants
RS-G-1.1	Occupational Radiation Protection
RS-G-1.2	Assessment of Occupational Exposure Due to Intakes of Radionuclides
RS-G-1.3	Assessment of Occupational Exposure Due to External Sources of Radiation

The radiological protection design of the UK HPR1000 has adopted the above codes and standards to implement the ALARP methodology, and the alignment of the UK HPR1000 radiological protection design with IRR17 will be presented using a consistency analysis to show how the UK HPR1000 radiological protection design fulfils the relevant requirements.

22.4 Regulatory Requirements and Safety Case

22.4.1 Radiological Protection Regulatory Requirements

The IRR17, made under the *Health and Safety at Work etc Act 1974 (UK) (HSW Act)*, Reference [5], implements the majority of the *Basic Safety Standard (BSS) Directive 2013/59/Euratom*, Reference [6], in Great Britain.

BSS reflects the recommendations of the ICRP, Reference [7], and lays down the basic safety standards for the protection of people from the effects of ionising radiation. The dose limits in IRR17 are presented in T-22.4-1 with their derivation from current ICRP recommendations.

T-22.4-1 Legislative Requirements

Recommendations/Legislation		Dose Limits
UK Legislation	IRR17	- Individual Worker: 20 mSv/yr. - Worker extremities and skin: 500 mSv/yr. - Worker lens of eye: 20 mSv/yr. - Public: 1 mSv/yr from all sources on a

Recommendations/Legislation		Dose Limits
		site. - Public extremities and skin: 50 mSv/yr. Public lens of eye: 15 mSv/yr.

The aim of IRR17 is to ensure that the exposure to ionising radiation from work activities is kept ALARP whilst complying with the specified dose limits. An employer working with ionising radiation is therefore required to take all necessary steps to restrict so far as is reasonably practicable the extent to which their employees and other persons are exposed to ionising radiation. The ICRP principles of justification, optimisation and limitation should be considered in order to demonstrate that an ALARP approach is being used.

22.4.2 Radiological Protection Safety Case

The strategy for demonstrating the radiological protection safety case is divided into three main parts:

a) UK safety requirements

The UK HPR1000 will be designed to meet the requirements of IRR17, including the dose limits listed in T-22.4-1.

b) RGP and initial assessments

Areas of good practice that are relevant to the activities and associated risks being assessed will be identified and justified. Initial assessments will be then performed against these examples of RGP.

c) ALARP assessment

ALARP assessments will be performed to ensure that doses received by workers and members of the public are ALARP.

In order to support the radiological protection safety case in an organised way, a strategy has been developed where the PCSR Chapter 22 is considered as a summary of safety case and the detailed information will be supported by a set of reference documents, primarily radiological protection topic reports, which describe where the arguments and evidence that substantiate the safety claims are presented. The topic reports cover source term, radiation zoning, shielding, demonstration for ensuring exposure is ALARP, worker dose evaluation, public dose evaluation from direct radiation, and radiation and contamination monitoring.

According to this strategy, the radiological protection safety case sets out the trail from claims through arguments to evidence as shown in T-22.4-2.

Radiological protection is a core discipline which links with a number of other disciplines, such as reactor chemistry, structural integrity, radioactive waste management, decommissioning, environment, fault studies and severe accidents. T-22.4-3 is the structure of the radiological protection safety case showing how the radiological protection safety claim links to other chapters.

T-22.4-2 Structure of Radiological Protection Safety Case

Claim	Sub-claim	Argument		Evidence		
	PCSR				Topic Report (TR)	
3.4.4: The risk to workers and members of the public from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP.	3.4.4.1: The risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP.	ALARP Methodology for Workers and Public Dose Assessment		PCSR: 22.6	TR 1: ALARP	
		1.1: Minimising Radiation Levels	Minimising Radioactive Source	PCSR: 22.5	TR 2: Source Term	TR 6: Worker Dose Evaluation
			Applying Radiological Protection Principles to Equipment, System, Layout and Operational Management	PCSR: 22.7	TR 1: ALARP	
			Radiation Zoning		TR 3: Zoning	
			Radiation Shielding		TR 4: Shielding	
			Radiation and Contamination Monitoring	PCSR: 22.8	TR 5: Monitoring	
		1.2: Minimising Exposure Time	Applying Radiological Protection Principles to Equipment, System, Layout and Operational Management	PCSR: 22.7	TR 1: ALARP	
			Radiation Zoning		TR 3: Zoning	
			Radiation Shielding		TR 4: Shielding	

<table>
<tr><th rowspan="2">Claim</th><th>Sub-claim</th><th colspan="2">Argument</th><th colspan="3">Evidence</th></tr>
<tr><th colspan="4">PCSR</th><th colspan="2">Topic Report (TR)</th></tr>
<tr><td rowspan="7"></td><td rowspan="4"></td><td></td><td>Radiation and Contamination Monitoring</td><td>PCSR: 22.8</td><td>TR 5: Monitoring</td><td rowspan="4"></td></tr>
<tr><td rowspan="3">1.3: Minimising Contamination Levels</td><td>Minimising Radioactive Source</td><td>PCSR: 22.5</td><td>TR 2: Source Term</td></tr>
<tr><td>Applying Radiological Protection Principles to Equipment, System, Layout and Operational Management</td><td rowspan="2">PCSR: 22.7</td><td>TR 1: ALARP</td></tr>
<tr><td>Contamination Zoning</td><td>TR 3: Zoning</td></tr>
<tr><td rowspan="2">3.4.4.2: The risk to members of the public from direct radiation during normal operation complies with UK legal requirements and is ALARP.</td><td rowspan="2">2.1: Minimising Radiation Levels</td><td>Minimising Radioactive Source (refers to PCSR 22.5)</td><td rowspan="2">PCSR: 22.10</td><td>TR 2: Source Term</td><td rowspan="2">TR 7: Public Dose from Direct Radiation Evaluation</td></tr>
<tr><td>Radiation Shielding (links to PCSR 22.7)</td><td>TR 4: Shielding</td></tr>
<tr><td colspan="6">An overall public dose assessment during normal operation is presented in PCER Chapter 7.</td></tr>
</table>

T-22.4-2 Structure of Radiological Protection Safety Case (Continued)

<table>
<tr><th rowspan="2">Claim</th><th>Sub-claim</th><th>Argument</th><th colspan="2">Evidence</th></tr>
<tr><th colspan="3">PCSR</th><th>Topic Report (TR)</th></tr>
<tr><td rowspan="2">3.4.5: The risk to workers and members of the public from the potential harmful effects of ionising radiation resulting from fault and accident conditions complies with UK legal requirements and is ALARP.</td><td>3.4.5.1: The risk to workers mitigating fault/accident conditions complies with UK legal requirements and is ALARP.</td><td>1.1: Considering all practicable measures to reduce exposure to radiation and contamination</td><td>PCSR:
22.7/22.8/22.11</td><td>TR 4: Shielding
TR 5: Monitoring
TR 8: Post-Accident Accessibility</td></tr>
<tr><td colspan="4">Dose assessment for other workers and members of the public for fault and accident analysis is presented in PCSR Chapter 12, Chapter 13 and Chapter 14.</td></tr>
</table>

Note:

PCSR Chapter 22: Radiological Protection	TR in relation to Radiological Protection
22.5: Source Term 22.6: Strategy for Ensuring Exposure is ALARP 22.7: Radiological Protection Measures 22.8: Plant Radiation Monitoring 22.9: Dose Assessment for Workers 22.10: Dose Assessment for Public form Direct Radiation 22.11: Post-Accident Accessibility (PCSR Chapter 12: Design Basis Condition Analysis) (PCSR Chapter 13: Design Extension Condition and Severe Accident Analysis) (PCSR Chapter 14: Probabilistic Safety Assessment)	1: Demonstration for Ensuring Exposure is ALARP 2: Radiological Protection Technical User Source Term 3: Radiation and Contamination Zoning 4: Radiation Shielding 5: Radiation and Contamination Monitoring 6: Worker Dose Evaluation 7: Public Dose Evaluation from Direct Radiation 8: Post-accident Accessibility Analysis

T-22.4-3 Radiological Protection Safety Case Route Map

Radiological Protection Claim	**Links to Other Chapters**
The risk to workers and members of the public from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP.	Link to system chapters: The radiological protection assessment sets the requirements on minimising dose through plant design and operations.
	Link to PCSR Chapter 31: The radiological protection assessment sets the requirements on minimising dose through plant design and operations.
	Link to PCSR Chapter 21: The minimisation of source term is demonstrated using the optimum chemistry regime.
	Link to PCSR Chapter 24: The decommissioning area requires a radiological protection input.
	Link to PCER Chapter 7: The public dose assessment requires a direct radiation assessment from PCSR Chapter 22.
The risk to workers and members of the public from the potential harmful effects of ionising radiation resulting from fault and accident conditions complies with UK	Link to system chapters: The radiological protection assessment sets requirements on minimising dose through plant design and states.
	Link to PCSR Chapter 31: The radiological protection assessment sets requirements

Radiological Protection Claim	Links to Other Chapters
legal requirements and is ALARP	on minimising dose through plant design and states.
	Link to PCSR Chapter 12, 13 and 14: The workers and members of the public risk assessment resulting from fault and accident conditions are predominantly covered in PCSR Chapters 12, 13 and 14, with the post-accident accessibility covered in PCSR Chapter 22.

Notes:

System chapters refer to PCSR Chapters 5, 6, 7, 10, 11, 17 and 23.

22.5 Source Term

22.5.1 Introduction

This sub-chapter describes the definition of UK HPR1000 radioactive sources and covers the various source terms for normal operation.

It describes the development of each source term and how the Technical Users utilise source terms for their specific areas, in particular how the Radiological Protection area uses source terms. For the other Technical Users, details on the source term and information on how each source term will be used will be provided in their respective chapters namely, *PCSR Chapter 24: Decommissioning*, *PCER Chapter 4: Radioactive Waste Management Arrangements*, Reference [8], and *PCER Chapter 6: Quantification of Discharges & Limits*, Reference [9].

The ALARP/Best Available Technique (BAT) considerations for source terms are described in this sub-chapter, while the ALARP/BAT assessments are described in others, notably *PCSR Chapter 17: Structural Integrity, Chapter 21: Reactor Chemistry, Chapter 31: Operational Management* and *PCER Chapter 3: Demonstration of BAT*, Reference [10].

The design basis accident source term, DEC-A source term and the severe accident source term can be found in *PCSR Chapter 12: Design Basis Condition Analysis and Chapter 13: Design Extension Conditions and Severe Accident Analysis*, respectively.

22.5.2 Overview of UK HPR1000 Source Term

The source term for the UK HPR1000 is defined as the types, quantities, physical and chemical forms of the radionuclides that have the potential to give rise to exposure to radiation, radioactive waste or discharges to the environment. The source term covers all aspects associated with the systems and processes within the UK HPR1000 that involve the production, transfer and accumulation of radioactivity.

The normal operation source term has been developed in a step by step, logical and iterative process.

Step 1: Define the approach/strategy and structure of the source term: *Normal Operation Source Term Strategy Report*, Reference [11], provides this information and will be continuously updated during step 3.

Step 2: Identify and justify the significant radionuclides for the UK HPR1000: *Report of Radionuclide Selection during Normal Operation*, Reference [12], provides this information and will be continuously updated during step 3.

Step 3: Define and justify the primary coolant source term: *Primary Coolant Source Term Methodology Report*, Reference [13], provides this information.

Step 4: Identify and justify the other normal operation source terms: dedicated supporting reports will be produced to provide this information.

Step 5: Undertake ALARP/BAT assessment to identify further opportunities to minimise the source term and implement them as relevant.

Step 6: Technical Users to use the ALARP/BAT source terms for their area assessment: Technical Users reports will provide this information.

The detailed definitions and justifications for each of the normal operation source terms are provided in the supporting reports and a summary is provided in this sub-chapter. The development and justification of the source terms are carried out using Operating Experience (OPEX) from Chinese and worldwide Pressurised Water Reactor (PWR) plants, which is described in the supporting reports.

22.5.2.1 Source Term Categories

There are seven categories of UK HPR1000 source terms, which are as follows:

a) Primary Coolant Source Term, which is the level of radioactivity within the primary coolant in the UK HPR1000. The primary coolant source term quantifies the concentration of each radionuclide present in the reactor coolant. The primary coolant source term is an input to most of the source terms defined below, such as the secondary coolant source term, derived source term and airborne activity;

b) Secondary Coolant Source Term, which is the level of radioactivity within the secondary circuit in the UK HPR1000. The secondary coolant source term quantifies the concentration of each radionuclide present in the secondary side liquid and steam;

c) Spent Fuel Assembly Source Term, which is the level of radioactivity within the spent fuel assemblies in the UK HPR1000. The spent fuel assembly source term quantifies the spent fuel gamma source strengths and neutron source strengths;

d) Derived Source Term, which is the level of radioactivity and activity deposited within each system in the UK HPR1000. The derived source term quantifies the concentration of each radionuclide present within the Structures, Systems and Components (SSC) (circuit pipes, ancillary equipment, etc.) and also quantifies the concentration of each radionuclide deposited on internal pipework, ancillary equipment and plant systems. It also quantifies the concentration of radionuclides in solid radioactive waste and in spent fuel pool water;

e) Gaseous and Liquid Discharges, comprising types, quantities, physical and chemical forms of the radionuclides within the monthly and annual radioactive discharges for gaseous and aqueous material. The gaseous and liquid discharges quantify the expected monthly and annual release of gaseous and aqueous material;

f) Airborne Activity, which is the concentration of airborne radioactive material in different buildings;

g) Activated Structures Source Term, which is the level of radioactivity within the reactor pressure vessel and internals, etc. The activated structure source term quantifies the concentration of radionuclides activated internally in the structure materials.

Further information is presented in *Normal Operation Source Term General Report*, Reference [14].

22.5.2.2 Radionuclide Selection

There is a large number of radionuclides generated in nuclear power plants, however many of these are commonly not considered as part of the source term for many different reasons, e.g. their half-life or their relevance for the technical users. Therefore, only appropriate radionuclides are considered in the normal operation source term.

The below listed aspects have been considered for radionuclide selection for the UK HPR1000:

a) Understanding of the origin of the radionuclides from within the core;

b) Identification, for each radionuclide group, of the sources and mechanisms for production of the radionuclides;

c) Identification, for each radionuclide group, of the full list of radionuclides which are to be produced based on theoretical analysis, literature research and OPEX;

d) Identification of physical and chemical characteristics for important/key radionuclides on the list;

e) Selection of relevant radionuclides and justification;

f) Management of the list of radionuclides.

There are four groups of radionuclides found in the UK HPR1000 and these are:

a) Fission products which are released from the fuel pins with cladding defects and can also be produced from tramp uranium;

b) Corrosion products which are generated from metal impurities within the coolant (from corrosion) as it passes through the core;

c) Activation products which are generated from the activation of the reactor coolant constituents, dissolved chemicals or entrained impurities (apart from corrosion products) as it passes through the core. Activation products can also be produced from activation of air constituents around the core or from non-fuel core components such as secondary neutron sources;

d) Actinide products which are produced from tramp uranium and which can also be released from the fuel pins with cladding defects.

Further information is presented in *Report of Radionuclide Selection during Normal Operation*, Reference [12].

22.5.2.3 Source Term Value Classifications

There are two sets of source term values for the UK HPR1000, which are as follows:

a) Realistic values, which give an average of the source term expected in the UK HPR1000 over a defined period of time. It can be used for areas such as disposability and dose impact assessments;

b) Design values, which give a conservative maximum value for the source term and are considered as the design basis for plant design (i.e. it is expected that this level would not be exceeded during operation, including during expected events). It serves notably as a basis for sizing the effluent treatment systems and shielding calculations.

They will be developed in the following way:

Source Term	**Realistic Source Term**	**Design Basis Source Term**
Primary Coolant Source Term	√	√
Secondary Coolant Source Term	√	√
Spent Fuel Assembly Source Term	–	√
Derived Source Term	√	√
Airborne Activity	√	√
Gaseous and Liquid Discharges	√	–
Activated Structures Source Term	√	√

22.5.2.4 Operating Modes Considered in the Source Term

Since different operating phases and physical and chemical conditions within the reactor can have a significant impact upon the source term, the UK HPR1000 source terms will be established based on a consideration of all operating modes of the fuel cycle during normal operation, including (but not limited to):

a) Start-up;

b) Power operation;

c) Shutdown;

d) Outage;

e) Expected events;

f) Other relevant and possible operating practices appropriate for consideration.

22.5.3 Primary Coolant Source Term

Primary coolant source term is defined as the types, quantities, physical and chemical forms of radionuclides in the primary coolant and primary circuit. It includes fission products, corrosion products, activation products and actinide products. These radionuclides are used for dose assessment, system design and radiological protection, etc. For the primary coolant source term, two sets of activity values have been selected to characterise normal operation conditions.

22.5.3.1 Fission Products

Fission products are produced from fuel and unavoidable tramp uranium. If the fuel cladding is damaged, the fission products in the fuel rods will be released into the primary coolant and migrate into the connected systems. According to the release mechanisms of the fission products to the coolant, only noble gases, including xenon and krypton, as well as volatile radionuclides such as iodine and caesium, will be released into the coolant.

UK HPR1000 fission products identification is based on theoretical analysis and relevant operating experience. The preliminary total number of fission product nuclides is 47, of which 43 have been selected using theoretical analysis of 700 fission products, and the remaining four fission products have been drawn from existing relevant OPEX. This list will be updated during step 3.

The UK HPR1000 list of selected fission products in the primary coolant is presented in T-22.5-1.

T-22.5-1 List of Fission Product Radionuclides

No.	Nuclide	No.	Nuclide	No.	Nuclide
1	Xe-133	17	Kr-83m	33	Te-127m
2	Kr-85	18	Cs-138	34	Ba-140
3	Xe-135	19	Mo-99	35	Sr-89

No.	Nuclide	No.	Nuclide	No.	Nuclide
4	Kr-88	20	Te-132	36	Sr-91
5	Xe-133m	21	Tc-99m	37	Sr-92
6	Xe-131m	22	Br-83	38	Te-125m
7	Cs-134	23	Br-84	39	Cs-132
8	I-133	24	I-130	40	Ba-139
9	Kr-85m	25	Te-134	41	Zr-95
10	I-131	26	Te-131m	42	Nb-95
11	I-135	27	Te-133m	43	Ce-141
12	Cs-137	28	Te-129m	44	Xe-138
13	Kr-87	29	Te-127	45	Ba-137m
14	Cs-136	30	Br-82	46	La-140
15	I-132	31	Rb-86	47	Sr-90
16	I-134	32	Te-129		

The detailed selection of fission products radionuclides is provided in *Report of Radionuclide Selection during Normal Operation*, Reference [12].

For steady-state conditions, the specific radioactivity of fission products is calculated based on theoretical analysis. Then, activity values are normalised respectively to { } GBq/t and { } GBq/t Dose Equivalent Iodine (DEI). The realistic source term ({ } GBq/t DEI) is representative of the average or best estimate values measured in the primary coolant of the Chinese Pressurised Reactor (CPR1000), French PWR and American PWR. The design basis source term ({ } GBq/t DEI) is the conservative value which is in line with almost all analysed data measurements obtained from CPR1000.

DEI is usually used to characterise the radioactive level in the primary coolant. It is defined as the I-131 equivalent concentration which alone could produce the same dose to the thyroid as the concentration of isotopic iodine mixture contained in the primary coolant.

The formula for calculating DEI is as follows:

$$DEI=A(I\text{-}131)+A(I\text{-}132)/30+A(I\text{-}133)/4+A(I\text{-}134)/50+A(I\text{-}135)/10$$

For transient conditions and shutdown conditions, the specific radioactivity of fission products is calculated based on steady-state values by multiplying a ‘peaking factor’ derived from OPEX. The ‘peaking factor’ is defined as the ratio of measured concentration in transient conditions to that in steady-state conditions.

The detailed analysis of fission products is presented in *Primary Source Term Methodology Report*, Reference [13].

22.5.3.2 Corrosion Products

Radioactive corrosion products are formed in two ways:

a) Corrosion of materials outside the core are transported by the reactor coolant into the reactor core, and subsequently activated; and

b) Activation of the constituent materials of the core itself followed by corrosion and/or release.

The type of corrosion products produced is closely influenced by the water chemistry condition and structure and equipment material selection.

Based on existing relevant OPEX, materials highly likely to be used in the UK HPR1000, characteristics of the identified radionuclides, UK and international practices as well as the identified radionuclides for Technical Users of the normal operation source term, the corrosion products selected as part of the UK HPR1000 normal operation source term are given below in T-22.5-2.

T-22.5-2 List of Corrosion Product Radionuclides

No.	**Nuclide**
1	Cr-51
2	Mn-54
3	Fe-59
4	Co-58
5	Co-60

No.	Nuclide
6	Ag-110m
7	Sb-122
8	Sb-124
9	Ni-63

Quantification of the corrosion products, which is based on recent OPEX, is presented in *Primary Source Term Methodology Report*, Reference [13].

22.5.3.3 Activation Products

Activation products are produced by neutron activation of the primary coolant constituents (e.g. Oxygen, Nitrogen and certain impurities) when it flows through the reactor core during normal operation or by neutron activation of constituents of non-fuel core components (e.g. Beryllium of secondary neutron sources) or air constituents (Argon-40).

22.5.3.3.1 Tritium

Tritium is a low penetration ability pure beta emitter radioisotope with a half-life of about 12.3 years. It is mainly produced by neutron activation of the soluble boron and lithium in the coolant, neutron activation of beryllium contained in the secondary neutron source rods and ternary fission reactions in the fuel and tramp uranium.

22.5.3.3.2 Carbon-14

Carbon-14 is a radioisotope with a long half-life (5730 years), and is a low-energy pure beta emitter. Carbon-14 in the primary coolant of UK HPR1000 is mainly produced by the neutron activation of Oxygen-17 (O-17) present from radiolysis of the coolant and the neutron activation of Nitrogen-14 (N-14) from nitrogen gas dissolved in the primary coolant.

22.5.3.3.3 Nitrogen-16 and Nitrogen-17

Nitrogen-16 (N-16) is mainly produced by the reaction O-16 (n, p) N-16. It has a short decay half-life of 7 seconds, emitting gamma rays with an average energy up to 6.15 MeV.

Nitrogen-17 (N-17) is mainly generated by the reaction O-17 (n, p) N-17. It has a half-life of 4 seconds and emits neutrons upon decay.

22.5.3.3.4 Sodium-24

Sodium-24 (Na-24) is produced by the reaction Na-23 (n, γ) Na-24 due to impurities in the coolant. It has a half-life of 15 hours, emitting gamma rays with energies of 1.37 MeV and 2.75MeV.

22.5.3.3.5 Argon-41

Argon-41 (Ar-41) is produced by neutron activation of Argon-40 (Ar-40) which is dissolved in the coolant and has a half-life of 1.83 hours. There is minimal argon dissolved in the coolant, as it is degassed before reactor start-up.

Based on existing relevant OPEX, characteristics of the identified radionuclides, UK and international practices as well as the identified radionuclides for Technical Users of the normal operation source term, the activation products preliminarily selected as part of the UK HPR1000 normal operation source term are presented in T-22.5-3.

T-22.5-3 List of Activation Product Radionuclides

No.	**Nuclide**
1	H-3
2	C-14
3	N-16
4	N-17
5	Na-24
6	Ar-41

The detailed analysis of activation products is presented in *Primary Coolant Source Term Methodology Report,* Reference [13].

22.5.3.4 Actinide Products

Uranium in the fuel can be converted into other actinides after being irradiated by the neutron flux in the core. Actinides are stable and will deposit on the surfaces of the fuel and circuit. The most prominent actinides include Uranium, Neptunium, Plutonium, Americium and Curium.

There are two sources of actinide products: one is from the tramp uranium present on the fuel despite the high quality of fuel manufacturing, and the other is from uranium released from severely failed fuel, which rarely happens. The tramp uranium composition constantly changes through the lifecycle of the UK HPR1000, and the initial composition is uncertain.

The detailed analysis of actinide products is presented in *Primary Source Term Methodology Report*, Reference [13].

22.5.4 Spent Fuel Assembly Source Term

Uranium in the new fuel will be converted into other actinides following neutron irradiation in the core, and fission products will be produced during unit operation. The structure materials in the fuel assemblies can be activated by neutrons in the core. Most of these radioactive nuclides can emit gamma rays and some can still emit neutrons after unit shutdown. These actinides, fission products and corrosion products in the fuel assembly material are considered under the radiological protection during fuel assembly transport and storage, and for the radioactive waste and spent fuel management, notably for the disposability assessment. The detailed analysis of spent fuel assembly source term will be presented in Spent Fuel Assembly Source Term Supporting Report.

22.5.5 Secondary Coolant Source Term

22.5.5.1 Definition of Secondary Coolant Source Term

Under normal conditions, the secondary circuit does not contain any radioactivity. It is completely segregated from the active Reactor Coolant System (RCP [RCS]) by the steam generators. Activity may enter the secondary circuit and connected systems through a leak in the steam generator. According to the radio-chemical specification, the operator can continue to operate the plant in the case of a small primary-to-secondary leak. Therefore, the secondary coolant source term can be calculated as the input for the potential radioactive dose evaluation for workers and members of the public in the event of a small primary-to-secondary leak.

Several systems are involved during the calculation of secondary coolant source term, including RCP [RCS], Steam Generator Blowdown System (APG [SGBS]), Main Steam System (VVP [MSS]) and Condenser Vacuum System (CVI [CVS]).

22.5.5.2 Radionuclide Selection

The radionuclides considered for the secondary coolant source term calculation include fission products, corrosion products and activation products. Some activation products such as N-16 and N-17 are not considered because the concentration of N-16 and N-17 in the secondary side will rapidly reduce to a negligible level due to their short respective half-lives. The alpha and beta particles emitted by radionuclides such as tritium or carbon-14 have sufficiently low penetration ability hence they can be sufficiently attenuated by containment structures such as piping and equipment. In terms of direct radiation, they make negligible contributions, but these are important sources for discharge, dose assessment and radioactive waste management.

The main radionuclides considered for each radionuclide group are as follows:

a) Iodine: I-131, I-132, I-133, I-134, I-135;

b) Noble gases: Kr-85, Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135, Xe-138;

c) Caesium: Cs-134, Cs-136, Cs-137, Cs-138;

d) Other fission products: Sr-89, Sr-90, Te-132, La-140, Ba-140;

e) Corrosion products: Cr-51, Mn-54, Fe-59, Co-58, Co-60, Sb-122, Sb-124, Ag-110m;

f) Tritium.

22.5.5.3 Secondary Coolant Source Term Calculation

The secondary coolant source term is derived from realistic and design basis levels of primary coolant source terms.

The detailed calculation parameters used for the calculation of the secondary coolant source term will be presented in Secondary Coolant Source Term Supporting Report.

22.5.6 Derived Source Term

22.5.6.1 Definition of Derived Source Term

A number of the SSC in the UK HPR1000 are significant in the determination and evolution of the source terms. The primary and secondary circuit and associated systems are the main contributors. The main SSC to be considered are:

a) RCP [RCS];

b) Chemical and Volume Control System (RCV [CVCS]);

c) Safety Injection System (RIS [SIS]);

d) Nuclear Sampling System (REN [NSS]);

e) Nuclear Island Vent and Drain System (RPE [VDS]);

f) Reactor Boron and Water Makeup System (REA [RBWMS]);

g) APG [SGBS];

h) Fuel Pool Cooling and Treatment System (PTR [FPCTS]);

i) Coolant Storage and Treatment System (TEP [CSTS]);

j) Solid Waste Treatment System (TES [SWTS]);

k) Gaseous Waste Treatment System (TEG [GWTS]);

l) Liquid Waste Treatment System (TEU [LWTS]);

m) Nuclear Island Liquid Waste Discharge System (TER [NLWDS]);

n) Conventional Island Liquid Waste Discharge System (SEL [LWDS (CI)]);

o) CVI [CVS];

p) Heating, Ventilation and Air Conditioning (HVAC), etc.

22.5.6.2 Radionuclide Selection

The radionuclides considered for the derived source term analysis include fission products, corrosion products and activation products. The concentration of N-16 and N-17 with half-lives of approximately 7 seconds and 4 seconds respectively, becomes negligible after a few minutes due to radioactive decay after shutdown and during outage. However, their contribution is not negligible during other operational phases where they must be considered. The alpha and beta particles emitted by radionuclides such as tritium or carbon-14 have sufficiently low penetration ability hence they can be sufficiently attenuated by containment structures such as piping and equipment. In terms of direct radiation, they make negligible contributions, but these are important sources for discharge, dose assessment and radioactive waste management.

The main radionuclides considered for each radionuclide group are presented below. This list may differ from one system to another and depends notably on the behaviour of radionuclides in the upstream systems

a) Iodine: I-131, I-132, I-133, I-134, I-135;

b) Noble gases: Kr-85, Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135, Xe-138;

c) Caesium: Cs-134, Cs-136, Cs-137, Cs-138;

d) Other fission products: Sr-89, Sr-90, Te-132, La-140, Ba-140;

e) Corrosion products: Cr-51, Mn-54, Fe-59, Co-58, Co-60, Sb-122, Sb-124, Ag-110m;

f) Activation products: N-16, N-17.

22.5.6.3 Derived Source Term Calculation

For each system, the specific activities of components are derived from the primary coolant source term. The components mainly consist of pipes, tanks, vessels, sumps, containers, filters, demineralizers and heat exchangers. For the RCP [RCS], the source terms for the steam generator, pressuriser and reactor coolant pump are also calculated.

The source terms are derived from realistic and design basis levels of primary coolant source terms. According to system and equipment design parameters, the derived source terms are developed in consideration of radionuclide migration in the coolant. Treatment, purification, degassing and demineralisation are considered and modelled in the calculation of derived source terms. In certain cases (e.g. the RCV [CVCS] filter), OPEX is used in order to obtain a more appropriate source term.

For systems in the reactor building, such as the RCP [RCS], RCV [CVCS] and REN [NSS], the shielding design source terms are based on the two plant conditions: full-power operation and shutdown. During full-power operation, N-16 and N-17 are the primary contributors to the gamma ray source term.

The detailed calculation parameters used for the calculation of the derived source term will be presented in Derived Source Term Supporting Report.

22.5.7 Gaseous and Liquid Discharges

The gaseous and liquid discharges from nuclear power plants during normal operation are used for the environmental impact assessment. The quantification of gaseous and liquid discharges and limits is addressed in *PCER Chapter 6: Quantification of Discharges & Limits*, Reference [9].

22.5.8 Airborne Activity

22.5.8.1 Definition of Airborne Activity

The airborne activity in the nuclear island buildings is calculated to estimate the resulting internal exposure to workers and external exposure caused by finite cloud shine.

22.5.8.2 Radionuclide Selection

The radionuclides considered mainly include noble gases and aerosols. Tritium is also considered as it is an important source term for inhalation and ingestion. N-16 and N-17 are not considered because the airborne concentration will rapidly reduce to a negligible level due to their short half-lives.

The main radionuclides considered for each radionuclide group are:

a) Noble gases: Kr-85, Kr-85m, Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135, Xe-138, Ar-41;

b) Corrosion products (aerosol): Cr-51, Mn-54, Fe-59, Co-58, Co-60, Sb-122, Sb-124, Ag-110m;

c) Activation products: tritium.

22.5.8.3 Airborne Activity Calculation

The airborne concentration of radiation nuclides is derived from realistic and design basis levels of primary coolant source terms.

Sources of airborne activity in the nuclear island buildings are:

a) Evaporation of radioactive liquid leaking from equipment or pipes;

b) Evaporation of pools, such as reactor pool, fuel pool and In-containment Refuelling Water Storage Tank (IRWST);

c) Activation of natural Argon-40 in the reactor cavity air.

The detailed calculation parameters used for the calculation of the airborne activity will be presented in Airborne Activity Supporting Report.

22.5.9 Activated Structures Source Term

Structure materials in the core, including the reactor pressure vessel and reactor vessel internals, can be activated by neutrons during normal operation. The main activated nuclides include Fe-55, Co-60 and Ni-63, along with other nuclides exhibiting a long half-life. The activity of these radionuclides in structural materials will gradually increase up to the saturation activity. Activated structures source term will not be released into the environment during normal operation and are considered as a part of decommissioning source term. The detailed analysis for the activated structures source term will be presented in Activated Structures Source Term Supporting Report.

Once the seven source terms have been calculated, they are reviewed through an ALARP/BAT process to identify if there are further reasonably practicable actions which could be considered for the UK HPR1000 to reduce the source term. This will focus on materials selection and treatment, chemistry regime and operating practices of the UK HPR1000. The aim is for the implementation of the safest configuration for the UK HPR1000 without incurring grossly disproportionate disbenefits in terms of time, cost and effort.

22.5.10 Technical Users Source Term

The Technical Users define the final levels of radioactivity considered for a particular assessment within a technical area of the safety and environmental case for the UK HPR1000. The Technical User source terms are made up of relevant parts of the primary coolant source term, secondary coolant source term, spent fuel assembly source term, derived source term, airborne activity and the activated structure source term. The following topic areas are the main users:

a) Radiological Protection

Uses a number of the source terms to determine shielding calculations and dose assessments. These are:

1) Primary coolant source term;

2) Secondary coolant source term;

3) Derived source term;

4) Spent fuel assembly source term;

5) Activated structures source term;

6) Airborne activity.

This information will be used to produce the Radiological Protection Technical User source term.

b) Radioactive Waste and Spent Fuel Management

This topic area will use a number of source terms to calculate the radioactive waste and spent fuel inventory which will be generated by the operation of the UK HPR1000. It will notably be used to develop the radioactive waste and spent fuel management strategy, design radioactive waste management systems and storage facilities and carry out disposability assessment. The source terms they will use are:

1) Primary coolant source term;

2) Secondary coolant source term;

3) Spent fuel assembly source term;

4) Derived source term.

This will be used to produce the Solid Radioactive Waste Management Technical User source term.

c) Decommissioning

The decommissioning Technical User source term will be based on the following source terms:

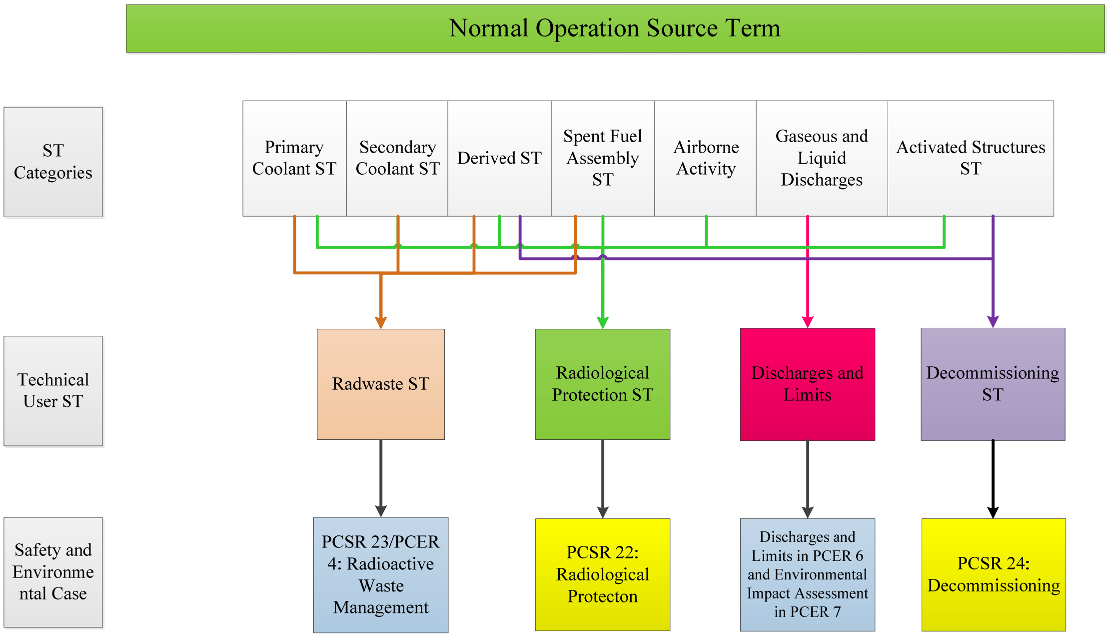
1) Derived source term;

2) Activated structures source term.

d) Environmental discharges

The gaseous and liquid discharges will serve as the discharges and limits.

The relationships between the normal operation source term and the Technical Users source terms are illustrated below in F-22.5-1.



F-22.5-1 Relationship between the Normal Operation Source Term and the Technical Users Source Term

Further information is presented in *Normal Operation Source Term General Report*, Reference [14] and the Technical User for Radiological Protection will be provided in Radiological Protection Technical User Source Term Report.

22.5.11 Accident Source Term

The design basis and DEC-A source terms, and the severe accident source term are addressed in PCSR Chapter 12 and Chapter 13 respectively.

22.5.12 Minimisation of Source Term

There are some key inputs which are considered at the start of developing the source term. These include defining the optimum chemistry regime and the optimum material selection through carrying out specific ALARP assessments in their topic areas. The source term takes the outputs from the ALARP assessments to characterise and quantify the types of activity which will be present in the UK HPR1000 primary circuit.

The minimisation of source term is demonstrated through:

a) Optimum chemistry regime;

b) Optimum material selection;

c) Optimum operating procedures.

22.5.12.1 Optimising Chemistry Regime

To minimise the source term, optimum reactor water controls are adopted as described in PCSR Chapter 21.

22.5.12.2 Optimising Material Selection

Material selection for systems and components containing reactor coolant is optimised in the design phase to minimise radiation exposure to ALARP. Materials are also selected to minimise corrosion products which are the primary radiation sources during shutdown maintenance. Material selection is described in PCSR Chapter 17.

22.5.12.3 Optimising Operating Procedures

Operating procedures can impact generation, accumulation and transfer of radioactive substances during operation. Optimisation of operating procedures is described in PCSR Chapter 31.

Further ALARP/BAT assessments will be carried out as needed to consider further potential and reasonably practicable actions source term reduction.

22.5.13 Management of the Source Term through GDA and Management of the Data

Source term development is very important for the UK HPR1000 GDA, affecting multiple areas, such as shielding and radiation zoning design, building layout, system and equipment design, dose impact assessments, decommissioning and disposability assessments, radiation monitoring and radio-chemical specification. Any design modifications affecting the source term should be registered and recorded appropriately. A feedback loop should be performed to ensure that the design modification is reflected in the source term data.

If design changes are identified, they will be carried out in accordance with the organisational arrangements and quality assurance arrangements provided in PCSR Chapter 20: MSQA and Safety Case Management. The proposed design changes will be confirmed after following the appropriate process, including ALARP and BAT assessments, and then communicated to all affected topic areas to enable them to undertake an impact assessment and subsequently update their respective safety or environmental cases.

22.6 Strategy for Ensuring Exposure is ALARP

22.6.1 Introduction

This sub-chapter describes the strategy to ensure that radiation exposure to workers and members of the public from direct radiation during normal operation is reduced ALARP. Normal operation includes start-up, power operation, shutdown, outage, maintenance and inspection. All buildings containing radioactive sources have been taken into account, which includes:

a) Reactor Building (BRX);

b) Nuclear Auxiliary Building (BNX);

c) Fuel Building (BFX);

d) Safeguard Building A (BSA);

e) Safeguard Building B (BSB);

f) Safeguard Building C (BSC);

g) Personnel Access Building (BPX);

h) Radioactive Waste Treatment Building (BWX);

i) Spent Fuel Interim Storage Facilities (BQF);

j) Interim Storage Facility for Intermediate Level Waste (BQZ);

k) Waste Auxiliary Building (BQS).

22.6.2 Radiological Protection Safety Requirements

The UK Health and Safety Executive (UK) (HSE) have developed a suite of guidance documents for ALARP in Reference [15], to explain how to understand the concept of ALARP and how to ensure the risk is reduced to ALARP. For radiological protection, this means demonstration that the doses received by workers in nuclear power plant and members of the public have been reduced to be ALARP.

IRR17 Regulation 9(2) requires that the hierarchy of control philosophy should be applied to restrict the exposure to ionising radiation. And priority should be given to improve the engineering control and design features.

Safety Assessment Principle (UK) (SAP), Reference [16], raised by Office for Nuclear Regulation (ONR) assign the Basic Safety Level (BSL) and Basic Safety Objective (BSO) for doses to individuals and groups, which correspond with the legal limit in IRR17. The radiation safety criteria are as shown in T-22.6-1.

T-22.6-1 Radiation Safety Criteria

Description	**Annual Effective Dose (mSv)**	
	BSL	**BSO**
Employees working with ionising radiation	20*	1
Other employees on the site	2	0.1
Average dose of any group on the site	10	0.5

Description	**Annual Effective Dose (mSv)**	
Any person off the site	1*	0.02
***: Legal Limit defined in Ionising Radiations Regulations, 2017**		

As part of the SAPs, RP.7 indicates that, to optimise the radiological protection, the hierarchy of control measures should be applied.

The ONR provides guidance, Reference [17], for the application of ALARP to new reactor designs in the UK. It suggests the inclusion of a detailed ALARP explanation and associated methodology. RGP is a basic requirement for ALARP demonstration, and risk/cost-benefit analysis can also be applied.

All employees on site and members of the public around should be considered in the dose assessment. The collective dose must be proved to be ALARP and forms a key element in evaluation of radiological protection.

For the UK HPR1000, the worker dose assessment, including the collective dose assessment, is described in Sub-chapter 22.9, while the public dose assessment from direct radiation is described in Sub-chapter 22.10.

22.6.3 Methodology for Ensuring Exposure is ALARP

For the UK HPR1000 design, all factors related to worker dose and public dose are taken into account to restrict the exposure of workers and members of the public to radiation, such as chemistry control, system design, equipment design, layout design, ventilation design, shielding design and so on, to ensure that the radiation exposure is ALARP.

The ALARP process for UK HPR1000 radiological protection is carried out using the following steps (as shown in F-22.6-1):

a) Selection of International RGP and OPEX

RGP forms a solid foundation for ALARP demonstration and is recognised by the ONR when applied appropriately.

RGP for the UK HPR1000 radiological protection is primarily drawn from existing codes and standards, (Nuclear) Industry Radiological Protection Co-ordination Group documents, previous GDA experience, and advanced PWR design features, such as:

1) Codes and standards

   - IRR17 and the Approved Code of Practice and guidance;
   - ICRP publications and IAEA standards, etc.

2) UK nuclear industry documents

   - The Application of ALARP to Radiological Risk;
   - Changeroom Design, Operation and Maintenance, etc.

3) Lessons learnt from previous GDAs

   - PCSR, PCER and their supporting documents for UK EPR, UK AP1000 and UK ABWR;
   - Regulatory Queries, Regulatory Observations and Regulatory Issues issued by ONR or Environment Agency during previous GDAs;
   - Assessment reports issued by ONR or EA for previous GDAs, etc.

4) Advanced PWR design features

   - Advance design features from the EPR and AP1000.

OPEX for the UK HPR1000 radiological protection is primarily drawn from existing PWR experience feedback and nuclear industry good practice, such as:

1) International OPEX sharing from authority websites, for example, UK Nuclear Institute, Organization for Economic Co-operation and Development (OECD) and Information System on Occupational Exposure (ISOE), etc.;
2) China General Nuclear Power Corporation (CGN) OPEX feedback.

b) Initial Evaluation

Based on information drawn from RGP and OPEX, items contributing to worker and public dose are identified. For the worker dose assessment, the tasks with high dose activities are identified, for example, maintenance for steam generators and fuel handling. For public dose assessment from direct radiation, the main target sources, such as the reactor core and steam generator, are identified.

c) Optioneering

During optioneering, the hierarchy of control philosophy is considered throughout the whole process.

The Eliminate, Reduce, Isolate, Control, Personal Protective Equipment (ERIC/PPE) methodology is applied when identifying the possible options to optimise the radiological protection, which means:

1) Elimination: eliminate the exposed activity or radiation source, to avoid the radiation risk;
2) Reduction: reduce the radiation source term or the frequency and duration of the exposed activity;

3) Isolation: isolate the radiation source or contamination from workers;

4) Control: administrative control such as temporary access controls to avoid unauthorised entrance or ventilation controls to prevent ingress of contamination;

5) Personal Protective Equipment (PPE): use appropriate personal protective equipment for radiological protection. PPE is not actually considered during design phases but is used during operation.

RGP and OPEX are the most important sources of options while applying ERIC/PPE methodology.

The identified options are then evaluated through a multi-disciplinary analysis according to the generic ALARP process. This will consider comments and feedback from operators and health physics, to determine the reasonable practicable options. More detailed information on option evaluation is provided in PCSR Chapter 33.

d) Implementation of the Reasonably Practicable Options

The reasonably practicable options are implemented to optimise the design and further reduce dose.

e) Risk Assessment

The dose assessments for the worker and for the public dose from direct radiation are carried out after the optimisation. The dose assessment results are then compared with the BSLs & BSOs to assess progress.

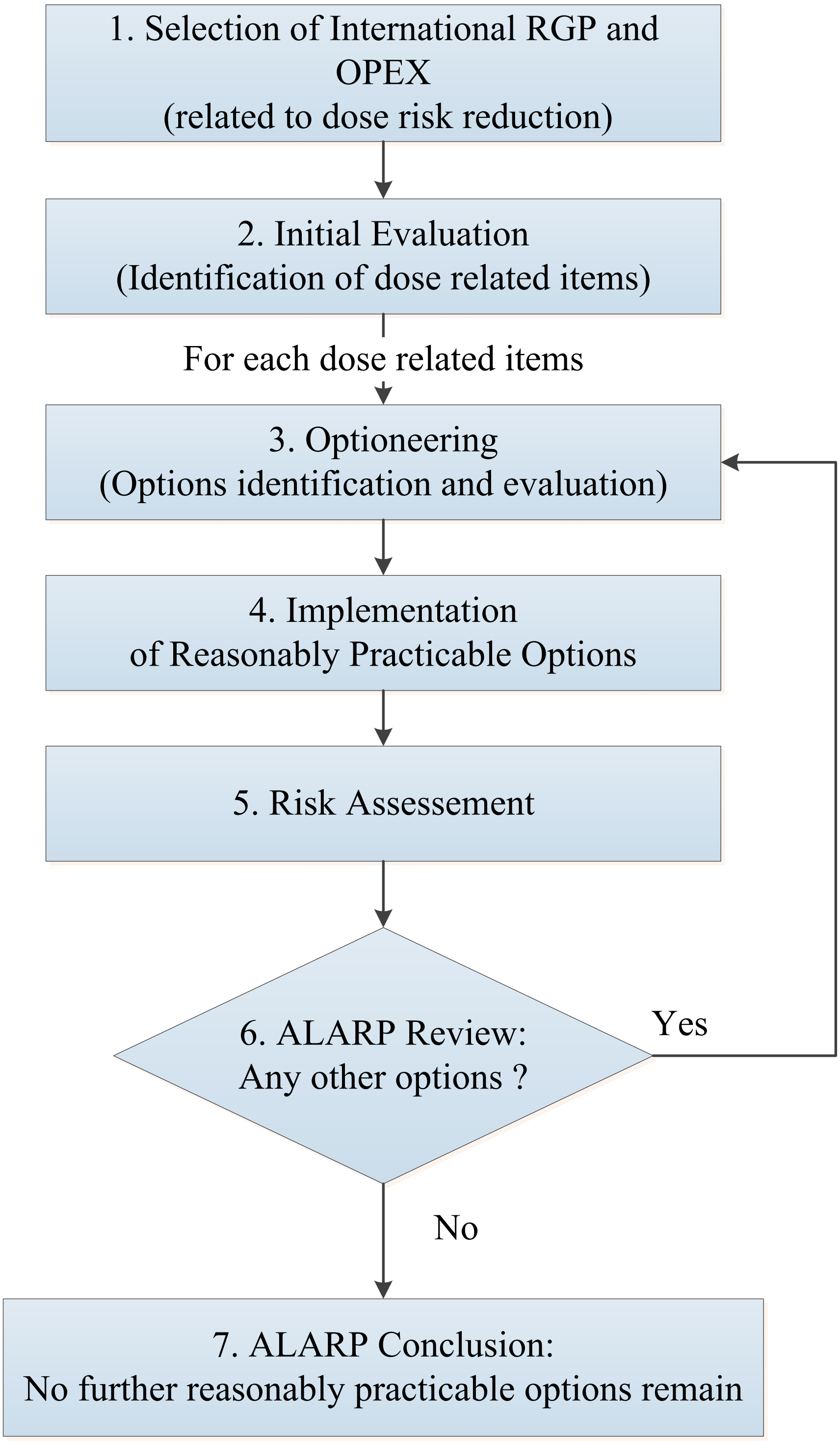
f) ALARP Review

The ALARP process is then reviewed to check whether:

1) The doses are below the BSL. If not, the ALARP process will be repeated to reduce the doses;

2) All the reasonably practicable options have been considered without any omission. If not, the ALARP process will be repeated to assess remaining options and further reduce the doses.

If both of the above have been achieved, an ALARP conclusion can be drawn.



F-22.6-1 ALARP Process for Radiological Protection

More details on the ALARP process for radiological protection and an example of a high dose task ALARP process will be provided in Demonstration for Ensuring Exposure is ALARP Topic Report.

22.7 Radiological Protection Measures

22.7.1 Introduction

This sub-chapter describes the radiological protection measures against direct radiation and contamination control for the UK HPR1000.

The generic design features for protection against direct radiation and contamination control are identified and listed based on the hierarchy of control, i.e. ERIC/PPE, methodology.

This sub-chapter summarises the specific design features that minimise direct radiation and contamination for different design areas. The relevant SSC for radiological protection are covered in the respective chapters on system, mechanical and civil engineering.

22.7.2 Radiological Protection Design Principles

22.7.2.1 Radiation Risks Assessment

In accordance with UK HPR1000 radiation safety requirements, various radiological protection measures shall be provided to control radiation risks so that no individual bears an unacceptable radiation risk and the dose to workers is kept ALARP. In order to achieve the above goals, the main radiation risks for workers shall be firstly identified. And then, the assessment and selection principle for suitable protection measures shall be provided.

Risk assessments shall be done prior to each task being undertaken. The tasks considered include operation, maintenance, fuel handling, in-service inspection, logistics, etc. For radiological protection design, risk assessments shall be carried out to identify the radiation hazards and to evaluate the risks to workers and members of the public.

Based on relevant worldwide OPEX, workers may be subjective to several kinds of radiation risks, including internal and external exposure. For external exposure, the main radioactive nuclides are the radioactive isotopes inside the coolant, resin and filter cartridge, concentrate, the deposited radioactive nuclide on the inner surface of equipment and pipes, surface contamination, and the neutrons and photons emitted from reactor core. The main source term for internal exposure is airborne activity generating from the leakage or evaporation of radioactive substances.

The types of radiation that need to be considered include: alpha particles, beta particles, gamma-rays, and neutrons. Bremsstrahlung may also need to be considered where high energy beta sources and large atomic number absorbers are present.

The types of radioactive nuclides that need to be considered include: fission, corrosion, activation and actinide products.

22.7.2.2 General Design Principles

The principles of ALARP and individual dose limitation are applied in accordance with UK safety requirements.

Two important aspects are considered in the general design to ensure that the occupational radiation exposure is ALARP:

a) Minimising the necessity for and the amount of time spent by the worker in radiation areas;

b) Minimising radiation and contamination level in daily service areas and areas adjacent to specific plant equipment.

The following aspects are considered during equipment and facility design to keep occupational radiation exposure ALARP during the plant operation:

a) Normal operation;

b) Maintenance;

c) Remote operation;

d) Refuelling and fuel storage;

e) In-service inspection;

f) Treatment of radioactive wastes;

g) Other anticipated operation events;

h) Decommissioning.

Improvements to equipment and facility design are identified based on the evaluation of OPEX from operating plants to reduce contribution of specific activities to collective dose. Radiological protection measures included in the UK HPR1000 design will benefit from developments and improvements made to PWRs across the world, all necessary measures to reduce radiation exposure will be implemented, if practicable.

Various factors, including maintenance type, duration and frequency of equipment, and radiation levels present during maintenance and other operations, are considered during the design of systems containing radioactivity to keep occupational radiation exposure ALARP.

22.7.3 Generic Considerations

Based on the above radiological protection principles, some detailed requirements are stated to guide the design, of which the key point is the hierarchy of control.

All reasonably practicable measures should be implemented for reducing exposure to radiation, including engineered safety measures and operational safety measures. Engineered safety measures include shielding, ventilation, containment, remote handling and interlocks. Operational safety measures, which may reduce exposure to the hazard during planned operations, include restrictions on occupancy, monitoring arrangements and alarms, the use of pre-determined and approved plans, and the use of appropriate barriers and signage.

During the design of the UK HPR1000, the hierarchy of control philosophy derived from IRR 17 Regulation 9(2) specifies the following order of preference for safety measures:

a) Passive engineered safety measures;

b) Automatically initiated active engineered safety measures;

c) Active engineered safety measures;

d) Administrative safety measures;

e) Mitigation safety measures.

Hence, passive engineered safety measures are preferred to active engineered safety measures and administrative safety measures.

An alternative hierarchy of protection, known as ERIC/PPE, is adopted alongside the above hierarchy of control to enable comprehensive consideration of safety measure types during radiological protection design.

22.7.4 Radiological Protection Design Features

22.7.4.1 Reactor Water Chemistry Regime

During normal operation, the water chemistry has a significant impact on radioactive deposition and levels of radioactive material in the RCP [RCS]. Therefore, the RCP [RCS] water chemistry of for UK HPR1000 will be designed and controlled under the optimum water chemistry state to control the generation and migration of radioactive nuclides. The two aspects needed to be controlled are:

a) Non-radioactive indicators in the primary coolant that have an effect on radionuclide generation and migration;

b) Radioactive indicators in the primary coolant during normal operation.

Non-radioactive indicators such as the hydrogen content, boron concentration, lithium concentration and the pH value of the primary coolant have a great influence on the generation and migration of radionuclides in the primary coolant. Therefore, these non-radioactive indicator control values have been considered in the design to maintain radiation exposure ALARP.

The concentration of the key radionuclides in the primary coolant shall be controlled during power operation to minimise the radiation exposure to workers and members of the public. The key radionuclides needed to be controlled are as follows:

a) I-131 dose equivalent in the primary coolant;

b) Total concentration of noble gases in the primary coolant;

c) Tritium concentration in primary coolant.

Detailed information on reactor water chemistry control can be found in PCSR Chapter 21.

22.7.4.2 Fluid System Design

The design considerations with regards to the fluid system for maintaining workers' radiation exposure to ALARP levels include the following:

a) Purification of radioactive substances;

b) Collection and containment of radioactive substances;

c) Minimisation of radiation-related manual operations.

22.7.4.2.1 Purification of Radioactive Substances

In order to maintain workers' radiation exposures ALARP, the fluid system design considerations for the purification of radioactive substances include the following：

a) For reactor coolant from the RCP [RCS], the fluid system shall be designed with radioactive purification devices;

b) For gaseous and liquid discharges released to the environment, the fluid system shall be designed with radioactive purification devices to reduce radioactive discharge to the environment;

c) For large open bodies of water inside the nuclear island, such as the fuel pool and reactor pool, radioactive purification devices shall be installed to reduce the radioactive concentration and meet the limits described in radiochemical specification;

d) The radioactive purification devices shall be designed by considering the physical and chemical forms of various radionuclides so as to ensure the expected decontamination efficiency for various radionuclides.

22.7.4.2.2 Collection and Containment of Radioactive Substances

In order to maintain workers' radiation exposures ALARP, the fluid system design considerations for the collection and containment of radioactive substances include the following:

a) Radioactive substances that leak in liquid form within the nuclear island should be collected and contained to prevent uncontrolled spread;

b) The fluid system design shall take all reasonably practicable measures to reduce the probability and magnitude of leakages of radioactive substances from the fluid system.

22.7.4.2.3 Minimisation of Radiation-related Manual Operations

In order to maintain workers' radiation exposures ALARP, the fluid system design considerations for minimisation of radiation-related manual operations include the following:

a) The fluid system design shall take all reasonably practicable measures to reduce the necessity for radiation-related manual operation;

b) If elimination of radiation-related manual operation is not reasonably practicable, the fluid system shall be designed to incorporate all reasonably practicable measures to ensure the workers operate in the lowest practicable radiation zone.

Further information on fluid system design can be found in PCSR Chapter 6, 7, 10 and 11.

22.7.4.3 Material Selection

Material selection of systems and components in contact with the primary coolant should be considered in the design to keep exposure of workers to radiation ALARP. The radioactivity of corrosion products depends primarily on the material used. Therefore, the selection or exclusion of certain materials or material constituents is one of the essential aspects of occupational radiation protection.

The material selection design aspects which are relevant to radionuclide generation include the following:

a) The easily activated nuclides present in material in contact with the primary coolant are released into the primary coolant due to corrosion or accidental events, and then form corrosion products as they pass through the core and become activated;

b) Readily activated nuclides present in the equipment material in the neutron-irradiated region inside and near the core form activation products following neutron adsorption.

In order to maintain radiation exposure ALARP, two types of measurement shall be taken for material selection. The first type is to control the release of harmful nuclides and the second type is to control the activation of materials inside and near the core.

For wear-resisting components, low quantities of cobalt base alloys are selected to meet the functional requirement of the components. Cobalt base alloys are used to manufacture very few parts of components, such as the Control Rod Drive

Mechanism (CRDM), reactor vessel internals, main pumps and selected valves.

Typical measures taken to control the release of harmful nuclides are as follows:

a) Limit the cobalt content of stainless steel in contact with the primary coolant. The cobalt content limits for typical components are as follows:

   1) Main components of the primary circuit:

      - Metal parts without direct neutron bombardment: { };
      - Metal parts under strong neutron flux: { };
      - Coatings and welding under neutron flux: { }.

   2) Non-primary main components (main metal parts, coatings and welds), but in the contact with primary coolant: { } as far as practicable.

   3) Due to the large inner surfaces of the steam generator tube bundle, the cobalt content of steam generator tube bundle shall be kept to { }.

b) Use of corrosion-resistant materials when selecting materials for components contained within radioactive systems;

c) Due to the risk of silver contamination, the gaskets of the primary coolant-related system equipment shall avoid using construction material with silver content, except for the reactor pressure vessel seal rings;

d) Due to the risk of zinc contamination, the potential risk of uncontrollable neutron poisons and activation products, gaskets for primary coolant-related system equipment shall avoid using construction material with zinc content;

e) Due to the risk of antimony contamination, antimony alloys shall be avoided for any component in the primary circuit. Antimony is controlled in the primary circuit materials as the residual element.

In order to control the activation of materials around the core, typical measures taken in material selection are as follows:

a) The cobalt content of the metals not exposed to the coolant but in the primary shield shall be no more than { } if practicable;

b) Metals in the secondary shielding of the reactor building shall contain lower cobalt content if practicable.

Further information on material selection can be found in PCSR Chapter 17.

22.7.4.4 Equipment Design

The following aspects are generally considered during equipment design to minimise the working time spent in radiation areas:

a) Reliability and durability of equipment to reduce or eliminate the repair and preventive maintenance requirements;

b) The convenience of planned maintenance or potential repair, including modular design of components, which enables ease of assembly, disassembly, and transfer of failed items to low radiation zones for repair;

c) Provision of redundant equipment to remove the need for immediate maintenance or repair should radiation levels prohibit worker entry;

d) Remote or mechanical means for operating, inspecting and repairing. Using remote monitoring or checking equipment, if practicable;

e) Installing permanent platforms around the steam generator eyeholes, eyelets, ports and access points as well as around the steam generators.

The following aspects are considered during equipment design to minimise the radiation level of areas close to equipment or components in need of attention:

a) Separating radiation sources and occupied areas wherever practicable;

b) Providing means to drain and flush areas and taking measures to remotely wash the contaminated equipment;

c) Considering methods for the minimisation of radioactive nuclide deposition during design of equipment, pipes, connections and valves;

d) Providing measures to prevent the spread of the contamination from operation areas (including direct connection to drains);

e) Separating highly radioactive equipment from less radioactive equipment, instruments and controls, wherever practicable;

f) Using high-quality valves, valve packing and gaskets to reduce leakage and spillage of radioactive nuclides;

g) Using coatings for the external surfaces of equipment to facilitate decontamination.

The corrosion products entrained in the coolant become activated as it passes through the core where they are exposed to a neutron flux. Additional attention is paid to reduce such products, assessing distribution and retention in the circuit. The measures considered are as follows:

a) Selecting corrosion resistant alloys and keeping the concentration of cobalt and other impurity nuclides as low as reasonably practicable;

b) Optimising the coolant chemical regime to reduce corrosion rate and control corrosion migration;

c) Optimising the surface condition and cleanliness of the pipes and equipment.

Further information on equipment design can be found in PCSR Chapter 17.

22.7.4.5 Designation of Areas

22.7.4.5.1 General Descriptions

The design of radiation zoning serves as a basis for the overall layout, ventilation system design, shielding design, as a measure to prevent the spread of radioactive contamination, to facilitate radiological protection management and occupational radiation exposure control, and to minimise the radiation doses to workers from direct and inhaled/digested dose.

During normal operation, according to the different levels of external radiation, surface contamination and airborne contamination, the radiation zoning for the UK HPR1000 is divided into undesignated and designated areas, and the designated area consists of supervised and controlled areas.

a) Undesignated area

   1) In this area, the annual dose of workers shall not exceed 1mSv. The effective dose rate shall be lower than 0.5 μSv/h. No airborne or surface contamination exists in this area.

b) Designated area

   1) Supervised area: this area is also known as the white area. In this area, the annual dose of workers may exceed 1mSv but shall not exceed 5 mSv. The effective dose rate shall be lower than 2.5 μSv/h. The beta surface contamination level shall be lower than 0.4 Bq/cm^2. In this area, special protection measures or safety measures may not be required, but the frequent supervision and evaluation of occupational radiation exposure conditions is required.

   2) Controlled area: in this area, the annual dose may exceed 5 mSv and the effective dose rate can be higher than 2.5 μSv/h. The beta surface contamination level may be higher than 0.4 Bq/cm^2. In this area, special protection measures or safety measures are required.

Workers entering the controlled area shall have received strict training and authorisation. Before entering the controlled area, workers will be provided with a passive dosimeter and a direct-reading electronic personal dosimeter and, if necessary, wear special protective equipment, and then enter the controlled area through barriers. When the tasks are completed, the workers have to pass through the portal contamination monitoring arranged in the sanitation passage before leaving the controlled area.

22.7.4.5.2 Radiation Classification for Compartments

According to the requirements in IRR17, a relatively large controlled area shall be divided into different sub-areas as needed for the convenience in management. Therefore, to facilitate radiological protection management and occupational radiation exposure control, according to the expected radiation levels and radioactive contamination levels, the controlled area is further divided into various sub-areas, i.e. green zone, yellow zone, orange zone and red zone.

a) Green zone: conventional working area

b) Yellow zone: intermittent working area

Protection measures shall be taken to monitor the dose rate and prevent the spread of contamination. In the presence of contamination, additional protective equipment needs to be used.

c) Orange zone: high radiation area

Access and presence within the orange zone shall be strictly controlled. Workers entering this area shall obtain the prior approval from qualified radiation protection managers. The radiation and contamination levels and the amount of time spent by the worker in radiation areas shall be assessed by the radiation protection managers before workers gain access. Additional personal protective equipment may be required in this area.

d) Red zone: extremely high radiation area

Normally, it is forbidden to enter the red zone. Workers entering this zone must be licensed in advance by the personnel in charge of units with special access permission. The access and the stay provisions for the orange zone are also applied to this zone. However, the exposure time is much more limited and careful preparations must be made for each operation. Special protective clothing or personal protective equipment may be required in this area.

During normal operation, especially for main equipment repair, the possibility of temporary or permanent rezoning of some areas shall be considered.

The dose rate limit values for radiation zoning are as follows:

T-22.7-1 Classification of Radiation Zone

Zoning	Dose Rate Limit	Notes
Green zone	≤10 μSv/h	Conventional working area

Zoning	Dose Rate Limit	Notes
Yellow zone	≤1 mSv/h	Intermittent working area
Orange zone	≤0.1 Sv/h	High radiation area
Red zone	>0.1 Sv/h	Extremely high radiation area

For further refinement of compartments in controlled area, the compartment radiation classification method and the letter-digit combined coding method are adopted. The letter indicates the order of magnitude of dose rate: ‘A’ corresponds to a dose rate of 10 μSv/h, ‘B’ corresponds to the dose rate of 100 μSv/h, and the rest may be deduced by analogy. The digit before the letter indicates the specific dose rate value under this order of magnitude of dose rate. For example, ‘2B’ indicates that the upper limit of dose rate is 200μSv/h. The classification of radiation zones for the UK HPR1000 is presented in T-22.7-2. Further information on the designation of areas will be provided in Radiation and Contamination Zoning Topic Report.

T-22.7-2 Detailed Classification of Radiation Zone

<table>
<tr><th>Classification</th><th>Dose Rate Limit</th><th>Zoning</th></tr>
<tr><td colspan="3">Undesignated area</td></tr>
<tr><td>---</td><td>≤0.0005 mSv/h</td><td>/</td></tr>
<tr><td colspan="3">Supervised Area</td></tr>
<tr><td>---</td><td>≤0.0025 mSv/h</td><td>White zone</td></tr>
<tr><td colspan="3">Controlled Area</td></tr>
<tr><td>A</td><td>≤0.01 mSv/h</td><td>Green zone</td></tr>
<tr><td>2.5A</td><td>≤0.025 mSv/h</td><td rowspan="4">Yellow zone</td></tr>
<tr><td>B</td><td>≤0.1 mSv/h</td></tr>
<tr><td>2B</td><td>≤0.2 mSv/h</td></tr>
<tr><td>C</td><td>≤1 mSv/h</td></tr>
<tr><td>2C</td><td>≤2 mSv/h</td><td>Orange zone</td></tr>
</table>

Classification	Dose Rate Limit	Zoning
D	≤10 mSv/h	
3D	≤30 mSv/h	
E	≤100 mSv/h	
3E	≤300 mSv/h	Red zone
F	≤1000 mSv/h	
3F	≤3000 mSv/h	
G	>3000 mSv/h	

22.7.4.5.3 Contamination Zoning

Contamination controlled areas are defined based on the potential for surface contamination and airborne contamination. T-22.7-3 presents the area designation for contamination in the UK HPR1000. It is developed based on the OPEX and current practice in CGN fleets. Good practice from the UK and other countries is also considered.

T-22.7-3 Designation of Areas for Contamination for UK HPR1000

Area Designation	Potential Contamination Level	Typical Area
C0	≤0.4 Bq/cm^2	Supervised area
C1	≤4 Bq/cm^2	Most area of controlled area
C2	>4 Bq/cm^2	Limited area of controlled area
C3	iodine and/or aerosol risk	---

Detailed information on contamination zoning will be provided in Radiation and Contamination Zoning Topic Report.

It shall be noted that, due to the existence of different source terms and the various needs for management, and detailed radiation and contamination zoning within the nuclear island, the operational stage may differ from that during the design stage of the UK HPR1000, although the radiation and contamination zoning principle is the same.

22.7.4.6 Ventilation

For the UK HPR1000, two types of measures are applied to reduce the risk of internal exposure. The first type is to minimise the leakage of radioactive substance, and the second type is to prevent the spread of airborne radioactive contamination. This sub-chapter mainly focuses on the latter.

The main function of the ventilation system is to support the physical containment through controlling and minimising the escape and spread of airborne contamination. The design features used to control the spread of airborne contamination are described as follows:

a) The gaseous radioactive effluents are collected before being discharged and the risk of leakage from the radioactive gas collection system will be minimised;

b) The concentration of aerosols and radioactive gases in the air is limited by ventilation;

c) Any movement of airborne radioactive contamination is from zones with lower to those with the higher potential contamination.

Before being discharged into the environment, the gaseous effluent is effectively filtered (by high efficiency filters and iodide absorbers).

Further information on ventilation system design can be found in PCSR Chapter 10.

22.7.4.7 Layout Design

To reduce the total duration of workers in radiation areas, the following aspects are considered during facility layout design:

a) Equipment, instruments and sampling points in need of daily maintenance, calibration, operation or inspection within radiation areas should be located at positions enabling ease of access and within lower dose rate;

b) Equipment emitting high dose rates should be operated, monitored or checked remotely or mechanically;

c) Means should be provided for transferring equipment or components in need of service to low radiation areas, if practicable.

The following aspects are considered for the generic facility layout design to minimise the radiation levels within personnel access areas and areas housing equipment required for manual operations:

a) Isolating the radiation sources and accessible operation areas if applicable, for example, pipes with high dose rates should not pass through operational areas;

b) Providing appropriate shielding between radiation sources and passages and operation areas;

c) Using permanent shielding to isolate equipment or components in operation areas, if applicable;

d) Adding temporary shielding to high dose section, for example, at points on the transfer tube where the dose rate exceeds the limit during spent fuel transfer;

e) Locating sampling positions, equipment and instruments in areas with low radiation levels;

f) Providing measures and appropriate space for mobile shielding when needed;

g) Providing decontamination measures for operation areas;

h) Providing measures for controlling contamination and for decontamination in contamination-prone areas.

22.7.4.8 Access Control of Controlled Area

22.7.4.8.1 Workers Access Control of Controlled Area

To avoid excessive exposure during unauthorised entry, appropriate access control for workers shall be installed at the entry and exit points of controlled areas.

The access control design features are as follows:

a) The controlled area access provided in the access building will be the only entrance and exit for workers (including operation, maintenance and visitors), excluding during emergency situations;

b) The access points and passages are located in areas with low radiation levels;

c) Portal contamination monitoring and small item contamination monitoring arranged at the access points ensures no uncontrolled radioactive contamination spreading;

d) Decontamination rooms will be set up in the controlled areas, and special decontamination supplies and appliances will be used for minor external contamination. Furthermore a dedicated medical office is situated outside the nuclear island to handle greater levels of body contamination.

22.7.4.8.2 Workers Access Control of Areas with High Exposure Risk

To avoid excessive exposure following unauthorised entry, appropriate access control shall be installed at the entry points of areas with high exposure risk inside controlled areas.

The access control design features are as follows:

a) Barriers are provided for high radiation level areas (orange zone and red zone) to restrict the access of unauthorised workers;

b) For areas with potential surface contamination, temporary or permanent surface contamination control zone can be established based on the contamination level. Workers shall require authorisation before gaining access and adequate monitoring equipment and personal protective equipment shall be provided.

22.7.4.8.3 Items Access Control of Controlled Area

In order to prevent the spread of radioactive contamination from a controlled area to a supervised area, strict control measures are required for items in the controlled area. The relevant design features are as follows:

a) The controlled area access provided in access building of the plant is the only entrance and exit for small items. It shall be designed to prevent the spread of radioactive contamination;

b) Surface contamination monitoring is arranged for potentially radioactive items exiting the controlled area;

c) Special access passages are to be set up for large contaminated items which are to be removed from controlled areas, and corresponding radioactive contamination monitoring methods and management measures are also established.

Detailed information on the access control of controlled area will be provided in Radiation and Contamination Zoning Topic Report.

22.7.4.9 Shielding

Shielding design is an effective measure to restrict dose during normal operation and fault and accident conditions. All necessary steps should be taken into consideration for shielding design to restrict so far as is reasonably practicable the extent of workers and members of the public which are exposed to ionising radiation.

The plant shielding design should be undertaken based on the radioactive source term during normal operation and fault and accident conditions. During normal operation, the main source terms considered are the radionuclides inside the radioactive equipment and pipes. During fault and accident conditions, the main radioactive sources considered are the containment sump activity, plateout activity and containment atmosphere activity.

The required parameters for shielding design are:

a) The geometry and nature of equipment and rooms;

b) The radioactive sources.

Depending on the complexity of the above parameters specifications, calculations and modelling may be required for a given configuration.

For the UK HPR1000, the general principles for the reference calculation point selection are as follows:

a) For the side source, the calculation point is located 30 cm from the surface of a wall or from equipment;

b) For the upper source, the calculation point is located 200 cm from the floor surface or the highest possible location;

c) For the lower source, the calculation point is located 60 cm from the floor surface;

d) For certain cases, the calculation point is located at the nearest point where the workers can reach due to consideration made to the existing arrangement of the equipment.

Typical materials used in shielding include lead, steel, water, and concrete. The material used for most of the plant shielding is ordinary concrete with a bulk density of approximately 2.35 g/cm^3.

The main calculation methods used in the radiation shielding design for the UK HPR1000 are point kernel method, discrete ordinates method and the Monte Carlo method.

Detailed information on shielding will be provided in Radiation Shielding Topic Report.

22.7.4.10 Radiation and Contamination Monitoring of Occupational Radiation Exposure

The radiation and contamination monitoring of occupational radiation exposure is presented in Sub-chapter 22.8.

22.8 Radiation and Contamination Monitoring

22.8.1 Introduction

Radiation and contamination monitoring is an essential approach to minimise the occupational dose. The purpose of this sub-chapter is to describe the design of plant radiation monitoring and radioactive contamination monitoring in the UK HPR1000.

This sub-chapter focuses on process radioactivity monitoring, effluent radioactivity monitoring, area gamma radiation monitoring, area airborne radioactivity monitoring, accident and post-accident monitoring, personal dose equivalent monitoring and radioactive contamination monitoring.

The process radioactivity monitoring, effluent radioactivity monitoring, area gamma radiation monitoring, area airborne radioactivity monitoring, accident monitoring and post-accident monitoring is performed by the Plant Radiation Monitoring System (KRT [PRMS]) and the function of personal dose equivalent monitoring and radioactive contamination monitoring is performed by the Controlled Area Access Monitoring System (KZC [CAAMS]).

22.8.2 Legislation and Standards

The KRT [PRMS] and KZC [CAAMS] are designed according to the standards produced by IAEA such as *Radiological Protection Aspects of Design for Nuclear Power Plants*, Reference [18], *Radiological Protection and Safety of Radiation Sources: International Basic Safety Standards*, Reference [19], *Safety of Nuclear Power Plants: Design*, Reference [20], *Radiological Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants*, Reference [21], UK legislation such as IRR17, and International Electrotechnical Commission (IEC) standards.

22.8.3 Function

22.8.3.1 Safety Function

The KRT [PRMS] is designed to monitor the integrity of barriers in order to confirm the containment of radioactive substances.

The barrier integrity is monitored by the KRT [PRMS] to provide information and data for the control of radioactivity-containing systems. The monitoring includes:

a) Primary coolant activity level monitoring of the REN [NSS] and RCV [CVCS] lines connected to primary coolant to detect fuel cladding failure;

b) Activity monitoring (mainly N-16 and noble gases) of main steam lines, of gaseous discharge from the CVI [CVS] and of steam generator blowdown water to detect the steam generator leakage or rupture;

c) Activity monitoring of cooling water in the Component Cooling Water System (RRI [CCWS]) so as to detect the reactor coolant system boundary failure.

The KZC [CAAMS] is not safety related.

22.8.3.2 Operational Function

The primary function of the KRT [PRMS] is to execute the following:

a) Identifying, with respect to time, abnormal changes of radioactivity levels on-site, so as to protect workers from unacceptable radiation exposure;

b) Continuously monitoring the radioactivity levels of the gaseous and liquid effluents, so as to ensure that the activity discharged from the nuclear power plant are lower than the required limits, and thus protect the environment and ensure that the radiation exposure of the public is ALARP;

c) Continuously monitoring the radiological levels of gases and liquids to check the integrity of barriers such as the fuel cladding and system pressure boundary, and to prevent the leakage and release of radioactive substances through the barrier;

d) Automatic initiation of specific alarms and related isolation devices when the monitored radioactivity level exceeds a certain specified threshold, so as to ensure that the radiation exposure of workers and members of the public is ALARP, and to protect the environment.

The KZC [CAAMS] executes the following four operational functions:

a) Measuring the dose rate received by workers and recording the accumulated dose during their stay in the controlled area;

b) Providing information on the assessment of individual dose and collective dose;

c) Monitoring the radioactive contamination levels of workers and small items when they exit the controlled area;

d) Preventing radioactive contamination from being taken out of the controlled area.

22.8.4 Process Radioactivity Monitoring

Process radioactivity monitoring is provided to monitor the radioactivity of process fluids and to detect the leakage from the steam generator, fuel failure, the integrity of the primary pressure boundary. Process radioactivity monitoring for the UK HPR1000 includes the following monitoring channels:

a) Monitoring of Steam Generator Leakage:

Steam generator leak can be detected by the KRT [PRMS] via:

1) Measurements of activity levels in the main steam system (these systems can also quantify the leak in litres per hour during power operation);

2) Measurements of activity levels in the steam generator blowdown water via the sampling circuit;

3) Measurements of the activity levels of non-condensable gas extracted from the CVI [CVS].

b) Monitoring of Fuel Cladding Failure:

These channels provide monitoring of primary coolant activity levels via REN [NSS] and RCV [CVCS] lines connected to primary coolant in order to detect fuel failures.

c) Monitoring Integrity of the Primary Pressure Boundary:

1) The gamma-activity measuring points in the three trains of the RRI [CCWS] (circuits theoretically not contaminated) are used to detect leakages of radioactivity containing systems;

2) The monitors arranged upstream and downstream of the RRI [CCWS] for the high-pressure RCV [CVCS] coolers are used to detect leakages on the high pressure coolers.

d) Monitoring of Gaseous Waste Treatment System:

The monitors arranged on the circulating pipeline and discharged pipeline of the TEG [GWTS] are used to monitor the radioactivity concentration of the purging pipeline and discharge pipeline, as a measure of TEG [GWTS] gaseous waste discharge control.

e) Monitoring of RPE [VDS] Relay Tank and Relay Sump:

Dedicated radiation monitoring channels are installed on RPE [VDS] process drains relay tank, the floor drains 1 relay sump and the floor drains 3 relay sump, to continuously monitor the radioactivity level and to perform isolation of discharge to downstream systems if high radiation levels are detected.

f) RCV [CVCS] Coolant Filter Monitoring:

Monitoring of RCV [CVCS] filters is used to continuously measure the dose rate of RCV [CVCS] filters to inform plant workers of the need for replacement prior to the dose rate reaching the predetermined threshold to ensure that the radiation shielding requirements of the solid waste treatment system (TES [SWTS]) are adhered to.

Further information on the above process monitoring channels will be described in Radiation and Contamination Monitoring Topic Report.

22.8.5 Effluent Radioactivity Monitoring

Gaseous radioactive waste is primarily discharged to the environment via the main stack while the liquid radioactive waste is discharged through the outfall. To indicate the radiation levels, to check compliance with the regulatory discharge limits and to avoid unplanned discharge events, effluent radioactivity monitoring is arranged. The effluent radioactivity monitoring consists of:

a) Radiation monitoring of gaseous discharge via the main stack;

b) Radiation monitoring of nuclear island liquid radioactive waste discharge;

c) Radiation monitoring conventional island liquid radioactive waste discharge.

Gaseous discharge monitoring is used to continuously monitor the radioactive concentration and total discharge activity of noble gases, aerosols and iodine discharged to the environment via the main stack. In addition to continuous monitoring, sampling of the noble gases, aerosols, iodine, tritium and carbon-14 is also undertaken and the samples are sent to the laboratory for analysis.

The nuclear island and conventional island liquid radioactive waste discharge monitoring is used to continuously measure the radioactive concentration (excluding tritium and carbon-14) of liquid waste released from the Nuclear Island Liquid Waste Discharge System (TER [NLWDS]) and the Conventional Island Liquid Waste Discharge System (SEL [LWDS (CI)]) to the environment, respectively. The monitors provide signals to stop the discharge pumps and isolate the discharge of liquid waste, and provide an alarm to the Main Control Room (MCR) if concentration exceeds a predetermined setpoint.

Further information on the radioactive discharge monitoring channels will be described in Radiation and Contamination Monitoring Topic Report.

22.8.6 Area Gamma Radiation Monitoring

Area gamma dose rate monitoring is used to measure the gamma dose rate in specified areas of the BRX, BNX, BFX, BSA, BSB, BSC and BWX, and to give an alarm when the dose rate exceeds the predetermined threshold, to protect workers from excessive external radiation exposure.

22.8.6.1 Installed Area Dose Rate Monitoring

Installed area dose rate monitoring is used to monitor the work place radiation levels during normal operation and fault and accident conditions. When the dose rates exceed the predetermined threshold, the audible and visual alarms will be activated to warn the plant workers of the radiation hazards.

During normal operation and fault and accident conditions, the area radiation monitoring shall be implemented for any area that has a direct influence on the plant workers and meets any of the following conditions:

a) Areas where the radiation dose rate may increase rapidly, and there is no other indicating device;

b) Areas where the radiation dose rate may increase to a level requiring evacuation of workers;

c) Areas where occasional high radiation dose rates may occur and have a direct influence on workers;

d) Areas where external operations by other workers may cause the rapid increase in dose rate;

e) Areas where there is access requirement under accident conditions.

Based on the above requirements, the following area radiation monitoring equipment is arranged in BRX, BNX, BFX, BSA, BSB, BSC and BWX to continuously monitor the dose rate levels and to provide radiation information for workers present or gaining access for operations, monitoring, inspection, patrols, maintenance or any other purpose. Installed area dose rate monitoring consists of the following monitoring channels:

a) Area gamma dose rate monitoring near reactor pool;

b) Area gamma dose rate monitoring on refuelling machine;

c) Area gamma dose rate monitoring near spent fuel pool;

d) Area gamma dose rate monitoring on spent fuel pool crane;

e) Area gamma dose rate monitoring in Fuel Handling and Storage System (PMC [FHSS]) transfer room;

f) Area gamma dose rate monitoring in filter replacement room;

g) Area gamma dose rate monitoring in hot laboratory;

h) Area gamma dose rate monitoring in sample analysis room;

i) Area gamma dose rate monitoring in hot mechanical workshop;

j) Area gamma dose rate monitoring in service corridors of safeguard building A;

k) Area gamma dose rate monitoring in service corridors of safeguard building B;

l) Area gamma dose rate monitoring in service corridors of safeguard building C;

m) Area gamma dose rate monitoring near personnel air lock in safeguard building.

Further information on area gamma monitoring channels will be described in Radiation and Contamination Monitoring Topic Report.

22.8.6.2 Portable Area Dose Rate Monitoring

Portable monitoring equipment is provided to measure the dose rate of equipment surfaces and local areas to provide information for radiological protection purposes. The monitoring is carried out periodically and/or as necessary according to the radiation monitoring programme.

When working in controlled areas, portable area monitoring equipment can be used to obtain dose rate information of a working area and, when a sudden increase of dose rate occurs, to warn workers to take appropriate actions (e.g. add temporary shielding, reduce the working time, or evacuate) to reduce the radiation exposure received.

22.8.7 Area Airborne Radioactivity Monitoring

Inhalation of contaminated air is the main cause of internal exposure for workers. To measure and record the radioactivity level of working areas. Installed airborne radioactivity monitoring equipment is arranged to provide information for workers and for dose assessments in the event of internal exposure.

Airborne radioactivity monitoring is mainly used to monitor the radioactive concentration of noble gases, iodine and aerosols within the BRX and ventilation system exhaust air, the radioactive concentration of aerosols within the hot laboratory and hot mechanical workshop exhaust air, and the gamma dose rate of the MCR intake air during normal operation and fault and accident condition, to provide information on radioactivity levels and to detect deviations which allow appropriate actions to be taken to avoid an increase in internal exposure or to minimise internal exposure.

22.8.7.1 Installed Airborne Radioactivity Monitoring

The airborne radioactivity monitoring can provide the workers and operators with the following information:

a) Accessibility of building areas;

b) The presence of leaks and the integrity of radioactive systems and equipment;

c) Habitability of the MCR;

d) Start-up signals of iodine filters in ventilation system;

e) Start-up and shutdown signals for the ventilation system; and

f) Discharge information for airborne radioactive substances under accident conditions.

Installed airborne radiation monitoring equipment is arranged in nuclear island buildings and in the BWX and consists of the following monitoring channels:

a) Airborne radioactivity monitoring for reactor building;

b) Radioactive concentration monitoring of noble gases in the Containment Sweeping and Blowndown Ventilation System (EBA [CSBVS]) large-flow exhaust air;

c) Radioactive concentration monitoring of noble gases in the EBA [CSBVS] low-flow exhaust air;

d) Aerosol sampling of EBA [CSBVS] low-flow exhaust air;

e) Iodine sampling of EBA [CSBVS] low-flow exhaust air;

f) Radioactive concentration monitoring of noble gases in exhaust air of the nuclear island ventilation system;

g) Radioactive concentration monitoring of aerosols in exhaust air of the hot laboratory;

h) Radioactive concentration monitoring of aerosols in exhaust air of the hot mechanical workshop;

i) Gamma dose rate monitoring of the MCR intake air;

j) Radioactive concentration monitoring in the BWX.

Further information on area airborne monitoring channels will be described in Radiation and Contamination Monitoring Topic Report.

22.8.7.2 Mobile Airborne Radioactivity Monitoring for Aerosol, Iodine and Noble Gas

Mobile airborne radioactivity monitoring is provided to measure the radioactive concentration of aerosols, iodine and noble gases in areas where temporary monitoring of airborne radioactive concentration is required and can be used in addition to installed monitoring channels.

Monitoring is carried out periodically as necessary according to the radiation monitoring programme, and the measurement results can be used to provide information for radiological protection measures as well as to determine the prohibited entry time for workers.

22.8.8 Accident and Post-Accident Monitoring

The main function of accident and post-accident monitoring is to provide workers with information on appropriate measures to be taken during and after accident conditions to minimise the impact on workers and members of the public. The accident and post-accident monitoring also helps to estimate the influence of radioactive substances released to the environment and to obtain information about the causes and progress of the accident. Accident and post-accident monitors include the following monitoring channels:

a) Accident and post-accident area gamma dose rate monitoring in the reactor building;

b) Accident and post-accident dose rate monitoring of noble gases in the containment annulus exhaust;

c) Accident and post-accident radioactive concentration monitoring of exhaust air released from containment;

d) Accident and post-accident dose rate monitoring of noble gases in the safeguard building exhaust;

e) Accident and post-accident radioactive concentration monitoring of noble gases in the stack during and after an accident;

f) Accident and post-accident radioactivity monitoring of EBA [CSBVS];

g) Accident and post-accident radioactivity monitoring of VVP [MSS];

h) Accident and post-accident radioactivity monitoring of APG [SGBS];

i) Gamma dose rate monitoring of MCR intake air.

Further information on accident and post-accident monitoring channels will be described in Radiation and Contamination Monitoring Topic Report.

22.8.9 Personal Dose Equivalent Monitoring

Portable personal dose rate monitoring equipment is provided in the entrance of controlled areas to measure the dose rate and to record the accumulated dose during their time in the controlled area. The measured data provides information on the assessment of individual dose and collective dose.

The objective of personal dose equivalent monitoring is to evaluate the workers' whole-body effective dose and the average equivalent dose for main irradiated organs or tissues by measuring the personal dose equivalent of external exposure and the personal dose equivalent of internal exposure. It helps to limit the dose equivalent and to ensure that the occupational dose of the workers does not exceed the dose limits.

Workers entering the controlled area should be provided with a passive dosimeter to record the personal dose equivalent. At the same time, the workers should also be provided with a direct-reading electronic personal dosimeter as a supplement to the passive dosimeter.

The personal dose equivalent of internal exposure is measured periodically by the whole body counter which is composed of radiation detector, gamma spectrometry analysis software, dose calculation software, and measurement controlling software and is used to measure and record the internal exposure dose.

22.8.10 Radioactive Contamination Monitoring

Installed contamination monitoring equipment including the portal monitors, hand and foot monitors and small items monitors are provided to monitor the radioactive contamination of the work clothing, body surface and small items of the workers when they exit the controlled area to prevent radioactive contamination from being taken out of the controlled area.

Portable contamination monitoring equipment is provided to monitor the surface contamination levels of workbenches, equipment, floors and walls in the controlled areas.

22.9 Dose Assessment for Workers

22.9.1 Introduction

This sub-chapter describes the worker dose assessment from radiation exposure during normal operation (including start-up, power operation, shutdown, outage, maintenance and inspection).

This sub-chapter summarises dose assessment for workers, which covers the collective dose evaluation and individual dose evaluation. The worker dose assessment will comply with radiation protection targets for normal operation corresponding to the numerical targets 1 and 2 in the SAPs. The evaluation of numerical target 1, normal operation - any person on the site, is presented in Sub-chapter 22.9.4; the evaluation of the numerical target 2, normal operation - any group on the site, is presented in Sub-chapter 22.9.3.

22.9.2 Radiological Protection Safety Requirements

Among the three fundamental principles recommended by ICRP 103, the following two principles are considered in worker dose assessment:

a) Optimisation: application of the As Low As Reasonably Achievable principle, moreover, for the purpose of optimisation of radiological protection, collective effective dose is an instrument for occupational radiation exposure optimisation, for comparing radiological technologies and protection procedures.

b) Limitation: the total dose to any individual should not exceed the appropriate limits specified by the commission.

Furthermore, for ensuring radiological protection in the design of new nuclear power plants, NS-G-1.13 describes the systematic applicable requirements of dose limitation and optimisation as a basis for the radiological protection measures.

The IRR17 requires protection of the health of workers and the general public against the hazards arising from ionising radiation.

The SAPs assign levels and objectives for radiation doses to individuals and groups; these are the BSLs and BSOs. The employer must ensure that doses to workers are ALARP in any case. The specific UK criteria for any employee and defined group of employees working with ionising radiation are presented in T-22.6-1.

22.9.3 Collective Dose

The process for collective dose evaluation is carried out based on the ALARP methodology and performed with the ALARP analysis process.

The collective dose evaluation for the UK HPR1000 consists of seven steps, which are as follows:

a) Selection of International OPEX

The UK HPR1000 design is an evolution of the CPR1000 and takes into account operating and design experiences, combining advanced facility design features.

Since HPR1000 (FCG3) is still under construction, its OPEX data are not yet available. Therefore, OPEX data selected for UK HPR1000 are taken from comparable stations across the world, for example, the CGN fleet and EDF fleet, if possible.

CGN has fifteen units of CPR1000 and four units of M310 in commercial operations in CGN. The collective dose evaluation will be based on OPEX data from these CGN units, using almost 120 reactor-years of data, in addition to consideration of EDF fleet OPEX data, if possible.

All the data from CGN units mentioned above are considered to set the reference dose which is a starting point for the UK HPR1000 collective dose evaluation.

b) Establishment of Initial Dose Database

The initial dose database is the basis for further optimisation and dose assessment, which is used to establish the reasonable reference dose for the UK HPR1000 collective dose evaluation.

The initial dose database mainly includes the following parameters:

1) Description of tasks related to dose activities;

2) Exposed doses for each task;

3) Exposed man-hours for each task.

c) Identification of High Dose Activities

The worker activities with the highest radiological risks are selected as representative examples to demonstrate that worker dose is ALARP. These are defined as the 'High Dose Activities'.

The following two aspects should be considered while identifying the high dose activities:

1) The single activity with high individual dose;

2) The most exposed group activities that contribute significantly to collective dose.

This collective dose was analysed and ranked using OPEX data to prioritise the worker activities with the highest radiological risks. The following activities are the top six high dose activities from the perspective of collective dose.

1) Works involving the reactor pressure vessel;

2) Works involving the steam generator;

3) Works involving the valve inspection and maintenance;

4) On-site service (including thermal insulation operation);

5) Waste processing;

6) In-service inspection.

d) Identification of Options for Risk Reduction Measures

Risk reduction measures are obtained by performing formal option identification with the aim of further reducing the worker dose. The hierarchy of control approach ERIC/PPE is used to identify the potential options throughout the UK HPR1000 project. The detailed optioneering is carried out base on the methodology described in Sub-chapter 22.6.

e) Selection of the Reasonably Practicable Options

Evaluation of the options identified by step d) is an important step in ALARP process, which aims to compare them with each other in order to make a decision on which can be selected as a reasonably practicable option.

Reasonably practicable options are confirmed by multi-disciplinary analysis:

1) Time-efficient methods to screen out the options that are clearly not reasonably practicable;

2) Detailed assessment of the remaining options to determine if they are reasonably practicable to implement.

Dose assessment is an important aspect of the detailed assessment. In this process, dose assessment primarily focuses on the parameter changes which are related to the local dose rate and operation requirements. More detailed information on option evaluation is provided in PCSR Chapter 33.

Evaluation of worker dose is performed when the reasonably practicable options have been identified. Based on RGP and the initial dose database, the effects of the optimisation scheme on source terms, exposure man-hours and dose rates are determined.

f) Collective Dose Assessment

The collective dose assessment will be carried out after optimisation. It is combined with revised doses for all tasks and classifies the worker groups with similar tasks.

g) Further Optimisation

Once the risk assessment has been undertaken and the reasonably practicable options are implemented, the next phase is to carry out a systematic review to look for any additional dose reduction measures that can be implemented, which is consistent with ALARP review in Sub-chapter 22.6.

22.9.4 Individual Dose

In IRR17, it is required that employers shall put arrangements in place to manage radiological protection for workers to keep the dose received by workers ALARP. A dose limit for employees working with ionising radiation is also implemented.

Based on the analysis of individual dose data from CGN fleets from 1994 to 2017, as shown in F-22.9-1, {

}. Therefore, it can be concluded that all CGN fleet meet the UK individual dose limit.

F-22.9-1 Distribution of Individual Dose in China

Furthermore, the UK HPR1000 is an evolutionary design of current PWR plants in China, with improvements on both the reduction of source term related to plant operation and maintenance, and of the exposed time by using the ALARP methodology. Therefore, the individual dose for UK HPR1000 will be lower than that on the operating Chinese plants.

Further information will be described in Worker Dose Evaluation Topic Report.

22.10 Dose Assessment for Public from Direct Radiation

22.10.1 Introduction

This sub-chapter describes public external dose assessment from direct radiation (including sky shine which is scattered radiation in the air) during normal operation (including start-up, power operation, shutdown, outage, maintenance and inspection).

An overall public dose assessment, including internal exposure from environmental discharge and external exposure from direct radiation can be found in PCER Chapter 7: Radiological Assessment. This sub-chapter focuses on the dose assessment of the public from direct radiation.

22.10.2 Radiological Protection Safety Requirements

The radiological protection safety requirements for members of the public are presented in Sub-chapter 22.6. The ONR sets numerical targets in its SAPs for nuclear facilities for evaluating whether radiological hazards are adequately controlled and risks are ALARP. Further information on ALARP can be found in Sub-chapter 22.6.

22.10.3 Dose Assessment for the Public from Direct Radiation

Based on the ALARP demonstration process mentioned in Sub-chapter 22.6, the method for dose assessment for the public from direct radiation is developed, which includes the following steps:

a) Identification of Main Radioactive Sources

In order to assess the impact of direct radiation from nuclear power plants to members of the public, the contribution of radioactive sources within the nuclear power plant to the public exposure should be fully considered. The main radioactive sources that have a significant impact on public direct radiation are shown in T-22.10-1. Radioactive sources that have negligible contributions to the public dose are excluded in the initial evaluation, these are generally located underground (as the soil provides significant shielding) or towards the inside of buildings and have significant shielding due to internal walls or ceilings (for structural or radiation zoning purposes).

T-22.10-1 Main Target Sources in Dose Evaluation for the Public from Direct Radiation

Building	**Main Target Source**
Reactor building	Reactor Core Steam Generators Main Coolant Lines
Nuclear auxiliary building	Coolant Storage Tanks
Fuel building	Fuel Pool Cooling and Treatment System Piping
Radioactive waste treatment building	Concentrate Tanks Spent Resin Storage Tanks

The dose assessment for the concept-design facilities, i.e. BQF, BQZ, BQS, will be carried out by the future licensee after the detailed design specification is completed.

b) Optioneering

This step is to make sure that the public dose from direct radiation has been reduced to be ALARP. The ERIC/PPE methodology is applied to identify the reasonably practicable options which could optimise the exposure for public from direct radiation.

It is first considered whether radiological risks can be completely eliminated or not. Elimination of the radiation sources is not reasonably practicable for the main target sources shown in T-22.10-1 because, for example, the reactor core, steam generators and main coolant lines are considered as target sources in case of reactor building assessment, which are essential to generate thermal energy. The coolant storage tanks are considered as one of the target sources in case of the nuclear auxiliary building assessment, which are necessary to receive reusable primary coolant or demineralised water (distillate) from the coolant treatment subsystem during normal operation. Elimination of exposure time is also not reasonably practicable because the occupancy for the public is not controlled and limited.

The risk reduction options are next considered because the elimination options are not reasonably practicable. The radioactive source minimisation and radiation shielding are considered as the best reduction options. Radioactive source minimisation for the UK HPR1000 is determined appropriately as a result of optioneering to ensure that all relevant risks including radiological risks are ALARP. The assessment of radiation shielding has demonstrated that the shielding is appropriately located and of adequate thickness to reduce doses. Further optimisation is not considered reasonably practicable as increased shielding thickness did not result in a significant reduction in the calculated exposure to the public.

According to the hierarchy of control philosophy, isolation, control and personal protective equipment should also be considered for further reducing public dose. However, the evaluation results indicate that these options are not reasonably practicable because any additional measures to isolate the target source from the public, further administrative controls or implementation of personal protective equipment for the public are inefficient, costly or unrealistic.

c) Implementation of the Reasonably Practicable Options

Reasonably practicable options are incorporated in the UK HPR1000 design. The minimisation of radioactive sources for the UK HPR1000 is summarised in Sub-chapter 22.5 and the specific design features adopted that minimise radiological risks for the UK HPR1000 are summarised in Sub-chapter 22.7.

d) Dose Assessment

The dose assessment for members of the public from direct radiation will be based on UK HPR1000 specific information and the occupancy pattern described in the *Initial Radiological Assessment Methodology*, Reference [22]. The dose assessment for public from direct radiation will be found in Public Dose from Direct Radiation Evaluation Topic Report.

e) ALARP Review

The ALARP process is finally reviewed to check whether there is any omission of other reasonably practicable option which can further reduce the public dose from direct radiation.

22.11 Post-Accident Accessibility

22.11.1 Introduction

This sub-chapter is to outline the regulatory requirements for radiological emergency, to identify the systems and components to which access is required in post-accident situations within the design considerations and to describe the methodology for assessing the dose of the emergency operation under accident conditions.

This sub-chapter identifies representative Design Basis Accident (DBA) and Severe Accident (SA) sequences for which direct intervention of workers is needed to mitigate the accident and to bring the plant to a safe condition. The dose received by the intervention workers during operation tasks will be assessed for the associated DBA and SA sequences.

The dose assessment for the workers who participate in the mitigation of accidents will be covered in this sub-chapter; while that for other workers in the buildings who may be accidentally exposed to radiation will be discussed in the numerical targets analysis. This sub-chapter only covers radiological protection issues under post-accident conditions. Information on emergency arrangements is given in PCSR Chapter 32.

The term ‘post-accident’ mentioned in this sub-chapter indicates the period from the initiation of a postulated accident to the period when the plant is returned to a safe condition (no more than 30 days). Medium and long term post-accident activities such as material/fuel recovery and clean-up operations are outside the scope of GDA.

22.11.2 Regulatory Requirements for Radiological Emergency

The definition and the conditions of intervention for a radiological emergency are specified in the statutory text of *Radiation (Emergency Preparedness and Public Information) Regulations (REPPIR)*, Reference [23]. Mitigation procedures are required for a radiological emergency.

Preliminary risk assessment is needed for emergency recovery actions development, including an estimation of potential dose uptake. Emergency dose levels may be authorised if necessary. Workers involved in emergency intervention may be exposed to doses in excess of the dose limits listed in IRR17. The emergency dose levels normally regarded as acceptable within UK regulations, Reference [24], are as follows:

a) Effective Dose: 100 mSv;

b) Equivalent Dose to Skin: 1000 mSv;

c) Equivalent Dose to Eye Lens: 300 mSv.

In the case of any dose assessment relating to the post-accident access requirements, confirmation will be needed to determine that the above requirements are satisfied. To be more specified, an area is accessible in post-accident situations for mitigation operations when the accessibility analysis results show the dose received by workers is strictly lower than the above limits. When the results show the dose received by workers is higher than the above limits, radiological protection measures must be enhanced to reduce this dose to a value lower than the limit.

For life saving actions, specific provision may be considered, in which case the Regulation 14(7) may take precedence over Regulations 14(2), 14(3) and 14(4) of REPPIR. However, it is desirable that for planning purposes the objective shall normally apply the following levels:

a) Whole Body Dose: 500 mGy;

b) Dose to Skin: 5000 mGy.

*(Doses quoted above in milligray are for deterministic effects)

In emergency exposure situations, the employer shall ensure that dosimetry is applied to the employees to assess the dose of ionising radiation received by an Approved Dosimetry Service that has been specifically approved by the UK regulator as able to assess accident doses. This is in line with Regulation 24 of IRR17, which states for any accident or other occurrence takes place which is likely to result in a person receiving an effective dose of ionising radiation in excess of 6 mSv, or an equivalent dose greater than 15 mSv for the lens of an eye or greater than 150 mSv for the skin or the extremities, the employer working with ionising radiation shall put into place reasonable provision to ensure that an appropriate dose assessment is made.

22.11.3 Post-Accident Access Facilities

The post-accident access requirements to SSC are determined according to the nature of the accident as well as the mitigation that is required in response to the postulated accident. The safety systems are described in PCSR Chapter 7.

For DBAs, the key facilities and systems that require direct intervention for accident mitigation operations are the MCR, PTR [FPCTS] and Emergency Feedwater System

[EFWS]. Most of the mitigation operations are completed remotely in the MCR.

For SAs, the key facilities and systems that require direct intervention for accident mitigation operations are the MCR, Extra Cooling System (ECS [ECS]), PTR [FPCTS] and Containment Filtration and Exhaust System (EUF [CFES]).

In identifying post-accident access requirements, two categories of post-accident access facilities are considered: areas and rooms where performed necessary actions bringing the plant to a safe condition within the safety analysis; areas and rooms leading to the above two areas and rooms. Systems that deliver the safety function required to mitigate the postulated DBAs and SAs and its supporting systems are both considered in the identification of post-accident access requirement.

For the mitigation operations completed in the MCR, the accessibility will be confirmed by the demonstration of habitability in MCR. The mitigation operations and associated access facilities will be determined in the Post-accident Accessibility Analysis Topic Report.

22.11.4 Methodology for Post-Accident Accessibility

The methodology for post-accident accessibility follows the following sequence:

a) Definition of representative accidents (DBAs and SAs) to be analysed;

b) Identification of SSC affected by the accident scenario where access is required;

c) Determination of source term of the relevant accidents;

d) Identification of access routes and access points for workers;

e) Determination of the parameters required in the access dose assessment;

f) Dose assessment for post-accident accessibility;

g) Comparison of the assessed dose with guidance limits to identify the lowest dose option strategy;

h) Production of maps to show post-accident access route clearly, including indication of predicted dose rates.

The details of post-accident accessibility will be described in the Post-accident Accessibility Analysis Topic Report.

22.12 Concluding Remarks

Based on the understanding of UK requirements, the radiological protection safety case for the UK HPR1000 is developed and summarised in this chapter, which is to support the claims that the risk to workers and members of the public from the potential harmful effects of ionising radiation complies with UK legal requirements and is ALARP.

The radiological protection safety case for the UK HPR1000 includes:

a) Identification of the radiological hazards associated with operation of the UK HPR1000;

b) Definition of the radioactive sources;

c) Definition of the strategy to ensure that the exposure to radiation is ALARP;

d) Identification and implementation of the specific design features based on the ALARP strategy;

e) Demonstration of radiation doses to workers and to members of the public complying with UK legal requirements and to be ALARP.

The definition of the UK HPR1000 design source terms and justification of the significant radionuclides are considered in this chapter and relevant supporting documents, which will be continuously improved during step 3.

The strategy to ensure workers and members of the public doses are ALARP and the radiological protection considerations applied to reactor water chemistry, fluid system design, material selection, equipment design, designation of areas, ventilation, layout design and radiation shielding are summarised in this chapter.

Detailed assessments of external and internal doses to on-site workers for key activities and that for the public from direct radiation including sky shine from all buildings and facilities containing radioactive sources are summarised in this chapter.

The radiological protection safety case is described together with the detailed arguments and evidence that substantiate them, supported by a set of reference documents, primarily radiological protection topic reports, which describe where the arguments and evidence that substantiate this safety case is presented. Where relevant, reference has also been made to other GDA PCSR chapters to ensure consistency across the whole safety case.

22.13 References

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