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UK HPR1000 GDA Project

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Preliminary Safety Report

Chapter 22

Radiological Protection

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22.1 List of Abbreviations and Acronyms

ALARP	As Low As Reasonably Practicable
APG	Steam Generator Blowdown System [SGBS]
BSL	Basic Safety Level
BSO	Basic Safety Objective
BSS	Basic Safety Standards
CGN	China General Nuclear Power Corporation
CPR1000	Chinese Pressurized Reactor
CVI	Condensate Vacuum System [CVS]
DEI	Dose Equivalent Iodine
GDA	Generic Design Assessment
HEPA	High Efficiency Particulate Air Filter
HPR1000 (FCG3)	Hua-long Pressurized Reactor under construction at Fangchenggang nuclear power plant unit 3
HSW Act	Health and Safety at Work etc Act 1974
HVAC	Heating, Ventilation and Air Conditioning System
ICRP	International Commission on Radiological Protection
IRR17	Ionising Radiations Regulations 2017
IRR99	Ionising Radiations Regulations 1999
IRWST	In-containment Refuelling Water Storage Tank
KRT	Plant Radiation Monitoring System [PRMS]
MSTM	Multi-Stud Tensioning Machine
NPP	Nuclear Power Plant
ONR	Office for Nuclear Regulation
OPEX	Operating Experience
ORE	Occupational Radiation Exposure
PWR	Pressurized Water Reactor
PTR	Fuel Pool Cooling and Treatment System [FPCTS]

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]
RGP	Relevant Good Practice
RIS	Safety Injection System [SIS]
RPE	Nuclear Island Vent and Drain System [VDS]
RRI	Component Cooling Water System [CCWS]
SAPs	Safety Assessment Principles
SEL	Conventional Island Liquid Waste Discharge System [LWDS(CI)]
TAG	Technical Assessment Guides
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]
UK HPR1000	The UK version of the Hua-long Pressurized Reactor

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

The objective of this chapter is to demonstrate that the design, and intended operation, of the UK version of the Hua-long Pressurized Reactor (UK HPR1000) will seek to ensure that during normal operation (operation, maintenance and inspection) and under fault or accident conditions, the radiation dose to workers is reduced As Low As Reasonably Practicable (ALARP). It contains:

a) An introduction to radiation protection legislation in the UK and China, and the importance of ALARP in the design and operation of Nuclear Power Plants (NPPs) in the UK.

b) How ALARP principles will be applied during design of the UK HPR1000.

c) An approach to estimating the source term for Hua-long Pressurized Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)) and the strategy to establish the UK HPR1000 source term.

d) Protection and provision against internal exposure and external exposure for HPR1000 (FCG3).

e) The design considerations of the on-site radiation monitoring for HPR1000 (FCG3).

f) A brief introduction to the HPR1000 (FCG3) dose assessment values and the methodology that will be used for optimisation of the collective dose in order to demonstrate that occupational radiation exposure for the UK HPR1000 design is ALARP.

g) A summary of radiological risk assessment under fault and accident conditions.

This chapter focuses on the radiation protection for workers, while the calculation method of dose assessment for members of the public is described in chapter 26.

This chapter will demonstrate:

a) That the Chinese regulations that the HPR1000 (FCG3) has been assessed against as well as UK guidelines and requirements for radiological protection are both derived from international recommendations.

b) That ALARP principles and the design considerations for ALARP will be implemented in the UK HPR1000 design.

c) That the source terms associated with radiation protection have been adequately considered.

d) That adequate radiation protection measures against exposure to radiation and radioactive substance will be provided during normal operation and fault or accident conditions.

e) That a proposed dose optimisation process aiming at reducing the potential doses received by workers to ALARP levels will be considered in UK HPR1000 design.

f) That the radiological risk fault and accident conditions will be adequately considered.

22.3 Radiation Protection Legislation

22.3.1 Regulatory Requirements

The Ionising Radiations Regulations 1999 (IRR99), Reference [1], made under the Health and Safety at Work etc Act 1974 (HSW Act), Reference [2], implement the majority of the Basic Safety Standards (BSS) Directive 96/29/Euratom, Reference [3], (recently superseded by 2013/59/Euratom, Reference [4]) in Great Britain. BSS reflects the recommendations of the International Commission on Radiological Protection (ICRP) and lays down basic safety standards for the protection of people from the effects of ionising radiation. The aim of IRR99 is therefore to ensure that exposure to ionising radiation from work activities is kept as low as reasonably practicable and does not exceed specified dose limits. The British (IRR99) and Chinese (GB 18871-2002), Reference [5] legislation are both derived from ICRP60, Reference [6] (recently superseded by ICRP103, Reference [7]) and the effective dose limits of UK HPR1000 will meet the requirements of IRR99. The effective dose limits of IRR99 are presented in T-22.3-1.

T-22.3-1 Effective Dose Limits of IRR99

UK Legislation	IRR99	Effective dose limits: - Individual Worker: 20 mSv/year. - Worker extremities and skin: 500 mSv/year. - Worker lens of eye: 150 mSv/year. - Public: 1 mSv/year from all sources on a site. - Public extremities and skin: 50 mSv/year. - Public lens of eye: 50 mSv/year.

EURATOM BSS 2013/59 is a high level document on European safety standards and it will be written into UK law by February 2018. Therefore the UK is currently writing a new set of Ionising Radiations Regulations to replace IRR99, which will be called the Ionising Radiations Regulations 2017 (IRR17). We refer to IRR99 at this stage and will review and implement any legislation changes at the next stage in Generic Design Assessment (GDA).

In addition to the legal limits defined in IRR99, the Office for Nuclear Regulation (ONR) has published a series of Safety Assessment Principles (SAPs) for Nuclear Facilities, Reference [8], together with supporting Technical Assessment Guides (TAG).

22.3.2 ALARP Principles

There is a requirement within IRR99 to demonstrate that all doses and potential risks from exposure to ionising radiation are as low as reasonably practicable throughout the life of the NPP from design onwards. In order to demonstrate an ALARP approach, an understanding is required of the risks involved and the measures that can be undertaken to reduce/avoid that risk.

'Reasonably practicable' in this case means that measures to reduce risk should be undertaken unless the sacrifice of doing so is 'grossly disproportionate' to the benefit.

When deciding whether a risk is ALARP, the most commonly used method is a comparison between the control measures proposed and Relevant Good Practice (RGP).

In situations where existing good practice does not fully reduce the risk to ALARP, a return to first principles may supplement good practice by comparing the remaining risk with the sacrifice required to further reduce it. It may be appropriate to perform first principles comparisons through the use of professional judgement, for example if the improvements are cheap to implement but the risk reduction is significant. For particularly complex or high risk areas, a more detailed analysis may be performed in which the risks and benefits are quantified and compared in some way to assist in a judgement over whether a risk has been reduced to ALARP.

ICRP26, Reference [9], introduces three principles to radiation protection: justification, optimisation and dose limitation. These principles have continued to be used in subsequent ICRP documents as the basis for a system of radiation protection and control of exposure to ionising radiation.

It is the principle of 'optimisation' that is directly relevant to ALARP. In order to effectively demonstrate ALARP in the early design stage of a new facility, such as the UK HPR1000, a process of dose optimisation is applied to areas of the plant design and how the design will perform throughout the entire lifecycle of the plant. This process involves:

a) Evaluation of the dose or risk of exposure for that particular work or set of operations.

b) Selection of an appropriate dose or risk criteria.

c) Identification of possible options for dose elimination, reduction or risk minimisation using an 'optioneering' process with decision recording.

d) Assessment and selection of the ALARP option by considering firstly the benefits and any detriments of each available option.

e) Implementation of the selected option including feedback loops to assess the success of the chosen option.

f) Application of review levels to ensure that risk to individuals is limited.

It is a minimum requirement that a new facility meets or ideally performs much better than the Basic Safety Levels (BSLs) in the SAPs, although it is recognised that this approach alone may not be considered ALARP. Set at levels significantly below the BSLs, the Basic Safety Objectives (BSOs) reflect improvements in nuclear safety standards and expectations over time. An optimisation process is therefore central in demonstrating ALARP below the BSLs, with the aim of meeting the BSOs where reasonably practicable for a new plant.

22.4 Design Considerations for ALARP

The general evolution of the UK HPR1000 design is discussed in chapter 2. The overall design objectives for the UK HPR1000 include a commitment to ensure that radiation exposures are ALARP.

Plant structures, systems and components will be designed to:

a) Optimise material selection in order to reduce radiation and/or contamination hazards so far as is reasonably practicable.

b) Reduce personnel access frequency and time spent in areas with radiation and/or contamination hazards so far as is reasonably practicable.

c) Provide suitable and sufficient protection against exposure to radiation and contamination in areas of the plant that require access during normal operation conditions and accident conditions.

d) Utilise a radiation protection programme that incorporates RGP for the safe management of radiation and contamination hazards on site, and throughout the lifetime of the plant, will strive to continually reduce the exposure of workers and members of the public to ionising radiation.

22.4.1 Facility Layout Design

The facility layout design of the UK HPR1000 will be developed with the aims of minimising radiation levels in areas of plant requiring access and reducing the amount of time personnel spend in those areas. Facility layout design considerations will include:

a) Simplification of routine operational and maintenance tasks. For example the HPR1000 (FCG3) has an integrated head package, which combines the control rod drive mechanism, lifting mechanism, gripping apparatus, cooling system of the control rod drive mechanism, lifting support, electrical and instrument system cables within an integrated part. The studs are tightened and untightened using an automated, remote operated Multi-Stud Tensioning Machine (MSTM). This simplifies the reactor top cover lifting process during refuelling operations, which significantly reduces the task duration and reduces the radiation exposure of personnel.

b) Physical separation of radiation sources and occupied areas where practicable. In the

HPR1000 (FCG3) fuel building, for example, there are separate areas for preparation and rinsing of spent fuel casks which reduces the radiation exposure to workers and reduces the potential for contamination spread. Further details on spent fuel management are provided in chapter 23.

c) Provision of adequate shielding between radiation sources and general access/service areas. An example is the HPR1000 (FCG3) nuclear auxiliary building, which houses the demineralisers and filters inside independent shielded cells which are designed to reduce radiation doses during maintenance.

d) Material surfaces that are easy to decontaminate with radial corners and edges (so far as reasonably practicable) to minimise the build-up of radioactive contamination. Where reasonably practicable, surfaces are coated in, or constructed from, a material that provides a continuous impermeable membrane that is resistant to damage. In the HPR1000 (FCG3) designated areas, there are requirements for surface coating to minimize the build-up of radioactive contamination.

e) Provision of adequate access controls between areas of lower and higher radioactivity. Well-defined (and if appropriate secure) boundaries will be set up between designated and undesignated areas and between any zones within the designated areas with appropriately sized and equipped interface points/change rooms, and the ability to alter zone restrictions within the designated area if required for operational reasons.

f) Zoning for the purposes of minimising radiation exposure and controlling contamination will be developed for the UK HPR1000 taking account of UK and Chinese good practice at existing NPPs or those currently undergoing GDA.

g) Access zones for areas containing equipment which could become contaminated are designed as labyrinths or fitted with shield doors in order to minimise their influence on adjacent rooms.

h) There is provision for adequate storage facilities for maintenance and logistics equipment (e.g. heat insulation and portable radiation shielding).

i) There is provision for an active workshop for the maintenance of contaminated equipment.

j) Provision of installed and portable radiation and contamination monitoring equipment, with well-defined monitoring routines and procedures. For example, personal contamination monitoring will be available at the interface points between areas of different radiological designations in order to control potential contamination spread.

22.4.2 Equipment Design

Equipment and component design considerations that will reduce the requirement for personnel access to radiation areas include:

a) The general design of high reliability, durable components and equipment to reduce or eliminate the requirement for repair or preventative maintenance.

b) Where reasonably practicable, the general design of easily removable, modular components or equipment that can be easily disassembled for maintenance or repair.

c) An important consideration in the selection of the materials used in the construction of components and equipment is the absence or low content of isotopes susceptible to activation to form radiologically significant radionuclides and thereby ensure potential radiation doses are ALARP. For example, the HPR1000 (FCG3) design reduces the amount of cobalt-based hard alloy in valves, and avoids the use of antimony alloys in the fabrication of pump bearings in and around the primary system.

d) Systems with the potential to accumulate contamination are designed with pipe work gradients to reduce the potential for radiation hotspots.

e) Areas with frequent worker activities such as equipment operation and sampling areas are mainly located in the areas with lower radiation risk in order to reduce radiation exposure of workers. Similarly, equipment control panels are mainly located in the areas with lower radiation risk to allow operation without access to higher dose rate areas and therefore ensuring the dose is ALARP.

f) Systems and equipment are designed to prevent contamination leaks through use of high quality components and high overall system integrity. However, in the event of a leak occurring, systems are designed to collect and control leaked material (e.g. through the use of bunds around vulnerable systems) to prevent contamination spread. Leaked material can then be transferred to a waste container and processed for disposal via a suitable waste route. Leak monitoring is covered further in chapter 17, and radioactive waste management in chapter 23.

g) As a general principle, components are designed with inspection in mind, with materials that reduce neutron activation and therefore dose to operators. An example is the design of insulation around the main pipelines in the primary system (until the second isolation valve) and in the secondary system of the HPR1000 (FCG3) which have quick installation/removal insulation layers to allow efficient inspection. This also includes shielding around components which can be removed efficiently to allow access.

h) System redundancy is an important aspect of plant operation and can ensure that the repair of failed components occurs when radiation and contamination levels are ALARP. Aspects of system redundancy are also explored in chapters 6 and 17.

i) Where it is practicable to employ, the use of remote techniques for operation, repair, monitoring and inspection of components and equipment allows doses to workers to be kept ALARP.

j) Optimisation with regard to waste management is implemented, e.g. remote operation/maintenance to reduce radiation exposure, ease of flushing storage tanks and pipework to remove hot spots of radiation.

22.4.3 Operational Considerations

Operational considerations at the design stage are focussed on the implementation of a robust radiation protection programme to ensure that radiation exposures are ALARP. This programme will incorporate RGP for the safe management of radiation and contamination hazards on site. Key aspects of this programme will include:

a) The use of prior risk assessments to assess the radiation and contamination hazards on site. Output from ionising radiation risk assessments will contribute to the overall methods used to restrict Occupational Radiation Exposure (ORE) to ALARP, design the radiation and contamination control systems, establish the facility layout and establish area designations and contingency plans.

b) Production of and adherence to site procedures (including local rules as defined in IRR99) for the safe management of radiation and contamination hazards.

c) Use of a radiation and contamination zoning regime which is kept under continuous review in order to segregate areas of higher and lower activity.

d) Use of installed and portable radiation and contamination monitoring systems for both area and personal use. These will include radiation dose rate measurements and surface/airborne contamination monitoring systems which will provide data on the accessibility of plant areas (including the control room), leak detection and system integrity.

e) A system to investigate any incidents involving radioactive material including any cases of suspected overexposure to ionising radiation.

f) Reduce internal exposure by isolation, ventilation, decontamination and use of protective clothing and respiratory protective devices.

g) Put clear unambiguous signs at entrances to avoid unnecessary exposure due to inadvertent entry.

22.5 Source Term

The aim of source term design is to recognise all radioactive sources and potential risks from exposure to ensure that the exposure of workers and members of the public are ALARP.

22.5.1 Approach to Estimating Source Terms

A preliminary introduction of the approach that has been used to estimate the source terms for the HPR1000 (FCG3) is given below.

The radioactive source term is used for the following purposes:

a) Evaluation of radioactive discharges during normal operations – to demonstrate compliance with the site dose constraint and to set discharge limits

    1) Airborne discharges

    2) Liquid discharges

    3) Solid waste produced

b) Radiation protection – design of shielding, zoning and building layout etc.

c) Design of waste treatment system, ventilation system and equipment

d) Collective dose assessment to workers during normal operation

e) Dose assessment to members of the public during normal operation

f) Disposability assessment

g) Design Basis Accident Analysis

    1) Evaluate dose to workers on site

    2) Evaluate dose to members of the public off site

The source term consists of the following main radionuclide groups:

a) Fission products: released from failed fuel or neutron reactions with tramp uranium

b) Activated corrosion products: produced by activation of metal elements used in alloys (or present as impurities) in the components of the primary circuit that have been released into the primary coolant by corrosion or erosion

c) Tritium: released as a ternary fission product or formed by activation of boron, lithium, or deuterium in the primary coolant

d) Carbon-14

    1) Formed by activation of Oxygen-17 in the primary coolant water or present in stainless steel structures

    2) Formed by activation of Nitrogen-14 in nitrogen gas dissolved in the primary coolant or Nitrogen-14 present in stainless steel structures

e) Nitrogen-16 and 17: formed by activation of Oxygen-16 and 17 in the primary coolant water

For normal operation condition, three types of source term have been developed termed ‘Realistic’, ‘Operation’ and ‘Design Basis’ source terms.

The source terms will be used as follows:

T-22.5-1 Source Terms Used for Each Purpose for HPR1000 (FCG3)

Purpose	Source Term
Evaluation of radioactive discharges from normal operations - compliance with site constraint	Operation source term
Evaluation of radioactive discharges from normal operations - setting discharge limits	Operation source term
Radiation protection - design of shielding, zoning and building layout etc.	Design Basis source term
Design of effluent treatment system, ventilation system and equipment	Design Basis source term
Collective dose assessment to workers during normal operation	Realistic source term
Dose assessment to members of the public during normal operation	Realistic source term & Operation source term
Disposability assessment	Realistic source term

The methodology used to derive each source term of the HPR1000 (FCG3) is outlined below.

22.5.2 Source Term Assessment Methodology

22.5.2.1 Core Inventory

Core inventory is an important input for the evaluation of radiological consequences in the safety analysis of reactor.

Core inventory can be obtained by burnup analysis code, which simulates the burning of the fuel assembly in the reactor. Since the results are used for safety analysis, the assumption of the calculation is conservative. Therefore, the results are ensured to be conservative.

22.5.2.2 Spent Fuel Assembly Source Term after Shutdown

In order to evaluate the radiation level during the operation, storage and transportation of spent fuel assembly, it’s important to calculate the gamma ray source strengths of the spent fuel assembly after shutdown.

The calculation method is same as that used for calculating core inventory. Since the results are used for shielding design, they should be conservative enough.

22.5.2.3 Primary Coolant Source Term

In general, the primary coolant source term is mainly used for system design and radiation protection. For the primary coolant, three types of activity values have been selected to characterise normal operating conditions:

a) The realistic source term is representative of the average values that may be measured in the primary coolant. It can be used for areas such as disposability assessment, collective dose assessment, routine discharges.

b) The operation source term is more conservative than the realistic source term. It is only used for maximum discharges.

c) The design basis source term is a conservative, or design basis, source term that assumes the design basis fuel defect level. It serves as a basis for sizing the effluent treatment systems and shielding requirements.

The primary coolant source term consists of several main radionuclide groups, which are described as follows.

22.5.2.3.1 Fission Products

The gaseous and solid fission products generated in the uranium oxide pellets will migrate into the gaps and reserve in the gaps between claddings and pellets during power operation. Once cladding failure occurs, fission products will be released into the primary coolant and migrate into the connected systems.

The specific activities of fission products are calculated, stated as 0.25 percent fuel defects, by code based on fission products migration mechanism model or formula, and are normalized to a reasonably conservative level.

Fission products source term consists of steady-state value and transient value. The steady-state activity can be estimated by code or differential equations. Transient activity value is obtained based on steady-state activity value by multiplying the peaking factor. The nuclide spectrum has been normalized to 0.1 GBq/t Dose Equivalent Iodine (DEI), 5 GBq/t DEI and 37 GBq/t DEI in steady-state operation according to the source term framework in China respectively. The typical radionuclide concentration of fission products source term is presented in T-22.5-2.

T-22.5-2 Fission Products Source Term

Nuclide	Realistic source term (GBq/t)		Operation source term (GBq/t)		Design basis source term (GBq/t)	
	Steady-state value	Transient value	Steady-state value	Transient value	Steady-state value	Transient value
Noble gases	3.10E+00	5.90E+00	1.68E+02	3.11E+02	1.25E+03	2.31E+03
Iodines	4.00E-01	5.00E+00	1.37E+01	1.77E+02	9.83E+01	1.28E+03
I-131eq	1.00E-01	1.90E+00	5.00E+00	9.92E+01	3.70E+01	7.36E+02
Cs-134	2.00E-03	9.90E-01	1.10E-01	5.40E+01	8.40E-01	4.00E+02
Cs-137	2.90E-03	1.20E+00	1.60E-01	6.60E+01	1.20E+00	4.90E+02

22.5.2.3.2 Activated Corrosion Products

Activated corrosion products are formed when corrosion materials are irradiated and activated by neutrons during their way through the core active zone. The main factors used to determine the specific activities of the activated corrosion products are the primary chemical conditions and the inner surface material of the primary coolant system. The specific activities of corrosion products are determined by Operating Experience (OPEX) and statistics of measurement data in comparable stations which have similar primary chemical conditions and inner surface material. Co-58 and Co-60 are the most important radionuclide in this category. The typical radionuclide concentrations of activated corrosion products source term is presented in T-22.5-3. Steady-state value and transient value are estimated by data statistics of measurement taken from the primary coolant water of 48 cycles in Chinese Pressurized Reactor (CPR1000), and shutdown value corresponds to the maximum of some cycles in CPR1000.

T-22.5-3 Activated Corrosion Products Source Terms

Nuclide	Activity concentration (GBq/t)		
	Steady-state value	Transient value	Shutdown value
Realistic source term			
Co-58	8.3E-03	5.6E-01	4.0E+02

Nuclide	Activity concentration (GBq/t)		
	Steady-state value	Transient value	Shutdown value
Co-60	7.6E-03	7.1E-01	2.5E+01
Design source term			
Co-58	5.7E-02	5.6E-01	4.0E+02
Co-60	5.1E-02	7.1E-01	2.5E+01

22.5.2.3.3 Tritium Source Term

Tritium is a radioisotope with a half-life about 12.3 years. The tritium generated in the primary coolant will be discharged into the environment. The tritium in the reactor coolant system is mainly from the following sources:

a) Neutron reactions with soluble boron and lithium in the reactor coolant.

b) Diffusion from the secondary neutron source rod.

c) Diffusion from the fuel rod, in which tritium originates from ternary fission.

Realistic value and conservative value of H-3 production could be estimated at 38.8 GBq/a per unit and 49.8 GBq/a per unit respectively.

22.5.2.3.4 Carbon-14 Source Term

Carbon-14 is a radioisotope with a long half-life (5730 years). It is a low-energy pure beta emitter (Emax=56 keV) generated in the Pressurized Water Reactor (PWR) primary coolant mainly from the following reactions:

a) Activation of Oxygen-17 in the primary coolant water or present in stainless steel structures.

b) Activation of Nitrogen-14 in nitrogen gas dissolved in the primary coolant or Nitrogen-14 present in stainless steel structures.

Realistic value and conservative value of C-14 production could be estimated theoretically at 306 GBq/a per unit and 435 GBq/a per unit respectively.

22.5.2.3.5 Nitrogen-16 and Nitrogen-17

Nitrogen-16 (N-16) is mainly produced by the reaction O-16 (n, p) N-16. It has a half-life of 7.13 seconds, and the average energy of gamma rays is up to 6.15 MeV. N-16 plays a major role in external radiation exposure and radiation protection during power operations. The activity of N-16 is evaluated considering the production with core flux irradiation and the reduction by decay when it flows through the primary loops.

Nitrogen-17 (N-17) is mainly produced by the reaction O-17 (n, p) N-17. It decays in a few seconds (half-life of 4.2 seconds) and emits neutrons. Its activity in the primary circuit has been established by the same method as N-16.

The activity of N-16 and N-17 leaving reactor vessel could be estimated theoretically about 5.37E+06 Bq/cm^3 and 1.76E+03 Bq/cm^3 respectively.

22.5.2.4 Secondary Coolant Source Term

The secondary coolant source term is an input for gaseous and liquid effluent release assessment, shielding and zoning design.

The secondary coolant source term is derived from realistic and design basis level of primary coolant source terms, which are normalized to equivalent iodine-131 of 0.1 GBq/t and 37 GBq/t respectively.

Steam generator tube defects cause the introduction of reactor coolant into the secondary cooling system. The resulting radionuclide concentrations in the secondary coolant depend upon the primary-to-secondary leak rate, the nuclide decay constant, and the steam generator blowdown rate.

22.5.2.5 Derived Source Term

Source terms of different systems in different buildings are used for radiation shielding and zoning design, dose assessment and equipment qualification in safety case.

Systems considered are as follows:

a) Reactor Coolant System (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) Steam Generator Blowdown System (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])

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

The derived source terms are derived from realistic and design basis level of primary coolant source terms, which are normalized to equivalent iodine-131 of 0.1 GBq/t and 37 GBq/t respectively. According to system and equipment design parameters, the derived source terms are developed in consideration of radionuclides migration in the coolant. Treatment, purification, degassing and demineralization are considered and modelled in the calculation of derived source terms. Conservative decontamination factors are used for filters and demineralizers. In certain cases (e.g. RCV [CVCS] filter), operating experience data are used in order to obtain a more appropriate source term.

For systems in the reactor building, such as RCP [RCS], RCV [CVCS] and REN [NSS], the shielding design source term are based on the two plant conditions normal: full-power operation and shutdown. During normal full-power operation condition, the contribution of nitrogen-16 is predominant in gamma ray source term.

The radioactive concentration of the derived source term is therefore based on the level of the primary coolant source term.

22.5.2.6 Gaseous and Liquid Radioactive Effluent Release

Gaseous and liquid effluent release from NPP during normal operation is an important input for environmental impact assessment. Gaseous and liquid effluent release is derived from realistic and operation level of primary coolant source terms, defined as expected value and conservative value, which are normalized to equivalent iodine-131 of 0.1 GBq/t and 5 GBq/t respectively.

For HPR1000 (FCG3), radionuclides produced during normal operation are released to environment in form of gaseous and liquid. The main atmospheric release pathways include TEG discharges, ventilation system discharges from the nuclear island building and condenser air removal system discharges resulting from primary to secondary leakage. Airborne effluents are normally released through the discharge stack after sampling. The main liquid release pathways include release of tritium discharge via TEP [CSTS], TEU [LWTS] discharges and secondary discharges resulting from steam generator blowdown processing and primary to secondary leakage. All liquid radionuclides are conducted to Nuclear Island Liquid Waste Discharge System (TER [NLWDS]) or Conventional Island Liquid Waste Discharge System (SEL [LWDS (CI)]) and will not be released to environment unless the sampling shows the concentration of radionuclide meet the discharge requirements. Radioactive decay is considered in the calculation of liquid effluent release.

Based on the different producing way of gaseous and liquid effluent and primary coolant source term, the gaseous and liquid effluent release is calculated by considering various gaseous and liquid discharge pathways, the physical model of ventilation system and waste treatment system physical model, radioactive decay, degassing, purification, filtration, and conservative decontamination factors.

The total annual releases of gaseous and liquid effluent are presented in T-22.5-4 and T-22.5-5.

T-22.5-4 Annual Releases of Gaseous Effluent (GBq per (year・unit))

Nuclide	Expected value	Conservative value
Noble gases	1.44E+03	7.87E+04
Iodines	1.09E-03	2.56E-01
H-3	3.88E+03	4.98E+03
C-14	2.94E+02	4.18E+02
Aerosol	2.27E-03	3.15E-02

T 22.5-5 Annual Releases of Liquid Effluent (GBq per (year・unit))

Nuclide	Expected value	Conservative value
Total (except H-3 & C-14)	1.02E+00	5.54E+00
H-3	3.49E+04	4.48E+04
C-14	1.22E+01	1.74E+01

22.5.2.7 Airborne Activity

The airborne activity in the nuclear island buildings is used to estimate the resulting internal exposure to workers and internal exposure caused by finite cloud shine. Airborne activity is also an important input for ventilation system design.

The airborne concentration of radiation nuclides is derived from realistic and design basis level of primary coolant source terms, which are normalized to equivalent iodine-131 of 0.1 GBq/t and 37 GBq/t respectively.

The airborne activity in nuclear island buildings is mainly from:

a) Evaporation of radioactive liquid due to leakage of 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 air of reactor cavity.

Based on the primary coolant source term, the airborne activity is calculated in consideration of the leakage of the components in nuclear island buildings, evaporation of pools, partition factor or the fraction of the leaking activity that is airborne, ventilation system design, free volume of nuclear island buildings and radioactive decay.

22.5.2.8 Accident Source Term

The source terms used for radiation protection design under accident condition are determined in consideration of the migration of radioactive fluid and possible changes of radiation types and levels.

22.5.3 Source Term Design for UK HPR1000

The strategy to establish the source term for UK HPR1000 will be based on the HPR1000 (FCG3). China General Nuclear Power Corporation (CGN) team has a good understanding of the UK requirements regarding source terms. In consequence, a gap analysis between the UK RGP and HPR1000 (FCG3) source term design will be performed. The source terms design for UK HPR1000 will be developed in consideration of the RGPs taken from other reactor safety cases, including those from previous reactor GDAs as well as OPEX from comparable NPPs across the world, which is related to HPR1000 (FCG3). According to the gap analysis results, a specific source term will be developed for UK HPR1000 during the GDA by modifying the HPR1000 (FCG3) source term design.

22.6 Radiation Protection Measures

The purpose of radiation protection is to provide adequate protection for workers and members of the public against ionising radiation during normal operation conditions and also under fault and accident conditions, and to ensure the radiological doses to workers and members of the public from nuclear facilities are ALARP. The dose limits and objectives for workers and members of the public are achieved by controlling internal and external exposure. The design of radiation protection for the UK HPR1000 will meet the relevant requirements specified in the SAPs.

22.6.1 Radiation Protection Classification and Zoning

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

Under normal operation, according to the different level of external radiation, surface contamination and airborne contamination, the radiation zoning for HPR1000 (FCG3) is divided into undesignated areas and designated areas, with designated areas consisting of supervised areas and controlled areas.

a) Undesignated area (HPR1000 (FCG3))

1) In this area, the annual dose of workers must not exceed 1mSv. The effective dose rate must be lower than 0.5 μSv/h.

b) Designated area (HPR1000 (FCG3))

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.

3) Supervised area: this area is also known as the white area. In this area, the annual dose of workers may exceed 1mSv but must not exceed 5 mSv. The effective dose rate must be lower than 2.5 μSv/h.

Workers entering the controlled area must receive area specific training and must be approved and controlled. Workers accessing controlled areas must go through the health physics facilities. Change rooms are provided for workers to change into work clothes or over clothes. These rooms are provided with lockers, wash sinks, showers and toilet facilities. As radiation workers exit the work areas, they go through body surface contamination monitors, shower for decontamination if needed, and receive radiologically controlled first-aid if needed.

Controlled areas are divided into different sub-zones to facilitate management, the radiological protection management and occupational radiation exposure control. For HPR1000 (FCG3), the controlled area is subdivided in 4 sub-zones indicated by colours as follows:

a) Green zone: sub areas corresponding to lowest dose rates (up to 10 μSv/h), no beaconing required.

b) Yellow zone: sub areas with controlled access, medium irradiation risk, special radiation protection dispositions are required. The maximum dose rate is 1 mSv/h.

c) Orange zone: these areas are exceptional access areas, with high irradiation risks. Any access requires special authorisation, with special radiation protection dispositions. The maximum dose rate is 100 mSv/h, or can rapidly increase to reach these values.

d) Red zone: access to these areas is usually forbidden. These areas are demarcated by a warning beacon, and are separated from other areas by physical barriers (e.g. walls, doors).

For HPR1000 (FCG3), the above 4 sub-zones are further subdivided according to the dose rate of rooms, known as the room radioactive classification. The room radioactive classification is applied to detailed shielding design (e.g. shielding design and equipment layout). This approach facilitates information exchange and communication between different organizations involved in the design work. The classification of radiation zones for HPR1000 (FCG3) is presented in T-22.6-1.

T-22.6-1 Classification of Radiation Zones in HPR1000 (FCG3)

Classification	Dose rate limit	Zoning
Undesignated area		
---	≤0.0005 mSv/h	/
Supervised Area		
---	≤0.0025 mSv/h	White zone
Controlled Area		
A	≤0.01 mSv/h	Green zone
2.5A	≤0.025 mSv/h	Yellow zone
B	≤0.1 mSv/h	
2B	≤0.2 mSv/h	
C	≤1 mSv/h	
2C	≤2 mSv/h	Orange zone
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	

The main purpose of radiation zoning under accident conditions is to define the dose rate level of the areas for which access is required, and to effectively guide the workers to intervene under accident conditions. Radiation zoning under accident conditions can also facilitate the radiation protection management and the occupational radiation exposure control during accident conditions, and to ensure the radiation safety of the workers. The radiation zoning under accident conditions is not the basis of access controls.

Under accident conditions, the radiation zoning boundary can refer to the radiation zoning boundary under normal operation, while the sub-zones can be further divided to guide the development of an emergency intervention plan.

22.6.2 Shielding Provision

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

The plant shielding design should be undertaken based on the radioactive source term under normal operation and fault or accident conditions. During normal operation, the main source terms considered are the radionuclide inside the radioactive equipment and pipes. During fault and accident conditions, the main source terms 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 geometry and nature of shield.

c) The radioactive sources.

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

For HPR1000 (FCG3), the general principles of the reference calculation point selection are as follows:

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

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 the nearest point where the workers can approach due to the arrangement of the equipment.

Materials used in shielding typically 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 methods used in the radiation shielding design for HPR1000 (FCG3) are point kernel method, discrete ordinates method and the Monte Carlo method.

22.6.3 Ventilation

For HPR1000 (FCG3), two types of measures are applied to reduce the risk of internal exposure. The first type is to minimise the leakage of radioactive substances, and the second type is to prevent the spread of radioactive contamination. A detailed description of minimising the leakage of radioactive substances is described in sub-chapter 22.4. This sub-chapter mainly focuses on the second type of measures.

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

a) The gaseous radioactive effluents are collected for discharge and leakage from the radioactive gas collection system is 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 the zone with the lower to that with the higher potential for contamination.

d) Before the gaseous effluent is discharged into the environment, the gaseous effluent is effectively filtered (high efficiency filter and iodide absorber).

22.6.4 Radiation and Contamination Monitoring of Occupational Exposure

Radiation and contamination monitoring of occupational exposure are essential approaches to minimise the occupational dose and to ensure that radiation exposures are ALARP.

For HPR1000 (FCG3), potential sources are measured by both installed and portable monitoring equipment. The installed equipment is provided at locations requiring frequent or continuous monitoring, with portable monitoring equipment being available to allow flexibility and also monitoring capability during any works.

Sufficient quantity of equipment is available, along with sufficient space for periodic inspection and maintenance. All equipment is periodically maintained.

22.6.4.1 Ambient Dose Equivalent Monitoring

One of the approaches to minimising occupational dose is to prevent unnecessary exposure by adequate radiation monitoring. From this perspective, installed area monitoring equipment at strategic locations (see 22.7.5) is provided to continuously measure, indicate and record the photon ambient dose equivalent rate at significant locations for radiation protection (e.g. main control room) to provide information regarding the plant's total radiological status. An alarm is activated upon detecting high radiation levels to warn workers to avoid inadvertent exposure.

Portable area monitoring equipment is provided to measure the ambient dose equivalent in any airspace. The equipment comprises gamma and neutron dose rate survey meters. The monitoring is carried out periodically and/or as necessary according to the radiation monitoring programme during normal operation conditions and also under accident conditions.

22.6.4.2 Radioactive Contamination Monitoring

Prevention and mitigation of radioactive contamination is also an essential approach to minimising the occupational dose. The airborne activity within any controlled areas is measured. In addition, monitoring of total air flow of Heating, Ventilating and Air Conditioning System (HVAC) at the discharge route (e.g. main stack) provides a sign of radioactive airborne contamination (see 22.7.5). The radioactive surface contamination of any systems, structures and components is measured by portable contamination monitoring equipment. Installed contamination monitoring equipment is provided to monitor the radioactive contamination on workers, and other items as they exit the controlled area to prevent exposure from the accidental spread of radioactive contamination to supervised areas. If the measured contamination level exceeds the prescribed level, measures are taken to control the spread of the radioactive contamination and to decontaminate.

22.6.4.3 Personal Dose Equivalent Monitoring

The personal dose equivalent from external exposure is continuously measured using personal dosimeters during any stay in the controlled area. All workers entering the controlled area must wear a passive integrating dosimeter and an electronic dosimeter on their work clothes. Also, the personal dose from internal exposure is estimated (e.g. based upon measured air concentrations and calculations). The total effective dose is then assessed using this information. Dose histories of all workers are recorded.

22.7 Plant Radiation Monitoring System

For HPR1000 (FCG3), the Plant Radiation Monitoring System (KRT [PRMS]) is designed to ensure the safe operation of nuclear reactor and prevent workers and members of the public from the excessive radiation exposure. The effective operation of KRT [PRMS] provides information on radioactivity levels and detects deviations which allow appropriate actions to be taken to avoid an increase in exposure or to minimise exposure. It can also provide support for the individual dose assessment as well as for the control of the radioactivity level caused by the postulated design basis accident.

22.7.1 Function

22.7.1.1 Safety Function

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

a) On-line continuous monitoring of fuel cladding failure.

b) Activity monitoring (mainly of N-16 and noble gases) of main steam line, gaseous discharge of Condensate Vacuum System (CVI [CVS]) and 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.

22.7.1.2 Operational Function

The KRT [PRMS] mainly executes the following four operational functions:

a) To identify in time the abnormal changes of radioactivity level on-site, so as to protect workers from the excessive radiation exposure.

b) To monitor continuously the radioactivity level of the liquid and gaseous effluents, so as to ensure that the activity of the radioactive substances discharged from the HPR1000 (FCG3) are lower than the limits, and thus protect the environment and ensure that the radiation exposure of members of the public is as low as reasonably practicable.

c) To monitor continuously the potentially radiological contaminated process fluid and atmosphere to check the integrity of barriers such as the fuel cladding, system pressure boundary, etc., and detect leakage helping to prevent release of radioactive substances through the barrier.

d) To start automatically some alarm devices 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 as low as reasonably practicable, and protect the environment.

22.7.2 Process Radioactivity Monitoring

One function of the process radioactivity monitoring subsystem is to measure the actual dose rate and/or activity concentration, to indicate any changes of the measured values and to indicate at an early stage any relevant escape of radioactive substances from activity-containing systems into systems that are normally free of activity so that appropriate corrective measures can be taken to stop such escape.

The other function of this monitoring subsystem is to measure the radiation level of the radioactivity-related system in order to detect any abnormal changes of the operating system and devices.

In doing so, workers are protected from unnecessary exposure to radiation and the release of radioactive substances can be minimised and balanced.

22.7.3 Effluent Radioactivity Monitoring

The main function of effluents monitoring is to monitor the gaseous and liquid discharges from NPP. The effluent radioactivity monitoring consists of:

a) Gaseous effluent radiation monitoring.

b) Nuclear island liquid waste effluent radiation monitoring.

c) Conventional island liquid waste effluent radiation monitoring.

The gaseous effluent monitoring is used to continuously monitor the radioactive concentration and total discharge activity of the noble gas, aerosol and iodine discharged to the atmosphere via the main stack. The nuclear island and conventional island liquid waste effluent radiation monitoring are used to continuously measure respectively the radioactive concentration (except for Tritium and Carbon-14) of liquid waste released from the TER [NLWDS] and SEL [LWDS (CI)] to the environment.

22.7.4 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 the accident to minimise the impact on workers and members of public. The accident and post-accident monitoring also helps to estimate the influences of radioactive substance released to the environment and obtain the information about the causes and progress of the accident.

22.7.5 Area Radiation Monitoring

The area radiation monitoring subsystem includes area gamma dose rate monitoring and area airborne radioactivity monitoring.

The area gamma dose rate monitoring is used to measure the gamma dose rate in the specified area of the reactor building, nuclear auxiliary building, fuel building, safeguard building and radioactive waste treatment building, and to give an alarm of the dose rate augmentation when the dose rate exceeds the predetermined threshold, to protect workers from excessive external radiation exposure.

The area airborne radiation monitoring channels are mainly used to monitor the radioactivity level of aerosol, iodine and noble gases in the air of ventilation system and the controlled areas of the reactor building, nuclear auxiliary building, fuel building, safeguard building and radioactive waste treatment building, and the radioactivity in the main control room intake air during normal operation and fault or accident condition, to protect workers in the controlled area from excessive internal radiation exposure.

22.8 Dose Assessment during Normal Operation

22.8.1 On-site Workers Dose Assessment

For HPR1000 (FCG3), the collective dose target value is 0.6 man-Sv per year per unit. Besides the individual dose target value for workers should meet the following requirements:

a) Annual effective dose of workers in the NPP controlled area should not be more than 15 mSv.

b) Annual effective dose of workers in the NPP supervised area should not be more than 5 mSv.

c) Annual effective dose of workers in the NPP non-radiation working areas should not be more than 1 mSv.

The optimised dose assessment for UK HPR1000 will be based on that of HPR1000 (FCG3).

22.8.2 Collective Dose Assessment

22.8.2.1 Introduction

Radiation exposures associated with operation of the UK HPR1000 design are primarily due to direct radiation from equipment and components containing radioactive material. In some areas of the plant, there is also the potential exposure due to airborne radionuclides.

Implementation of a radiation protection programme in line with UK good practice which has associated controls on surface contamination levels, will provide confidence that radiation exposure from surface contamination is considered to be negligible in the calculation of expected collective radiation dose.

22.8.2.2 Dose Optimisation Methodology

A dose optimisation process with the aim of reducing the potential doses received by workers to ALARP levels will be taken into consideration.

The majority of the dose optimisation process will be analysis of operational feedback from several PWR NPPs currently operating in China. Operational feedback includes annual collective dose data, task specific data where available, and information on good practice developed on site. The plant designers and engineers will be central to the dose optimisation process.

Analysis of the OPEX data will be undertaken during the GDA process to identify tasks that will be optimised for the design and result in lower collective doses for workers. Any predicted dose reductions will be accompanied by justifications for the optimisation step.

A review of the radiation protection aspects against international good practice will be completed to ensure that RGP is captured and included in the dose optimisation process. Optioneering assessments will be performed where required in order to assist in demonstrating that doses for particular operations are ALARP.

The evolution in reactor design and associated improvements to collective dose reduction will be demonstrated. Summation of the optimised task-based collective dose data will allow a collective dose target for UK HPR1000 to be developed that demonstrates ALARP principles.

22.8.3 Off-site Members of the Public Dose Assessment

The dose assessment for members of the public form nuclear facilities is described in chapter 26.

22.9 Radiological Risk Assessment under Fault and Accident Conditions

The purpose of this sub-chapter is to outline the dose and risk assessment under fault and accident conditions.

22.9.1 On-site Radiological Risk Assessment for Design Basis Accident

Release of radioactive material during design basis accident may cause additional radiological consequence and risk to any person on-site. Accident analysis and workers dose assessment under design basis accident will be taken into consideration for on-site radiological risk assessment in order to demonstrate that the radiological risk is adequately controlled and to justify the overall risks are ALARP.

22.9.2 Off-site Radiological Risk Assessment for Design Basis Accident

Release of radioactive material during design basis accident may cause ionising radiation and radiological risk to any person off-site. Source term, weather conditions, potential exposure individual and site-specific parameters will be taken into consideration for off-site radiological risk assessment in order to demonstrate that the radiological risk is adequately controlled and to justify the overall risks are ALARP.

22.9.3 Radiological Risk Assessment under Design Extension Condition

Radiological consequence for design extension condition may be greater than that for design basis accident. Radiological risk assessment will be conducted for both workers and members of the public under design extension condition. In addition, since severe accident may cause significant consequence in the scope of area or country, total risk of 100 or more fatalities, either immediate or eventual will be assessed to demonstrate that the radiological risk is adequately controlled and to justify the overall risks are ALARP.

22.10 References

[1] UK Statutory Instrument 1999 No. 3232, The Ionising Radiations Regulations 1999, ISBN 0-11-085614-7, 1999.

[2] UK General Public Act, The Health and Safety at Work etc., Act 1974, 1974

[3] Euratom, Basic Safety Standards Directive 96/29/Euratom (superseded), 1996.

[4] Euratom, Basic Safety Standards Directive 2013/59/Euratom, 2013.

[5] GB, National Standard of P.R.C., Basic Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. GB 18871, 2002 edition, October 2002.

[6] ICRP, Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, Ann. ICRP 21 (1-3) (superseded), 1990.

[7] ICRP, Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4), 2007.

[8] ONR, Safety Assessment Principles for Nuclear Facilities, Rev 0, 2014.

[9] ICRP, Recommendations of the International Commission on Radiological Protection, ICRP Publication 26, Ann. ICRP 1 (3) (superseded), 1977.