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Title

Chapter 29: Quantification of Radioactive Effluent Discharges and Proposed Limits

Executive Summary

This report is Chapter 29 of the Environment, Safety, Security and Safeguards (E3S) Case for the Rolls-Royce Small Modular Reactor (RR SMR). This Chapter provides an assessment of potential discharges of aqueous and gaseous radioactive effluent from the RR SMR to the environment under normal operating conditions, based on the RR SMR at Preliminary Concept Design (PCD) stage, and published operating experience (OPEX) data from Pressurised Water Reactors (PWR). The Chapter includes the underlying methodologies, including underlying assumptions and parameter values used to calculate the preliminary quantification of radioactive discharges to the environment.

Annual discharge limits are proposed for the predicted discharges to the environment, based on the interim quantification of discharges across the full fuel cycle, and identification of 'significant radionuclides'. Justification of the proposed allowance for uncertainty and variability of disposals based on the estimated worst case annual plant discharge (WCPD) is provided.

The annual discharges of aqueous radioactive effluent predicted to arise from the RR SMR, normalised to 1GWe are mostly found to be below the average of reported or forecasted discharges from comparable PWRs. The normalised annual discharge of gaseous radioactive effluent from the RR SMR are consistent with the average of forecasted or reported discharges from other PWR plants, except for the discharges of 'other radionuclides' which is higher than the reported values. This is likely an artefact of the conservative approach used at the preliminary stage to estimate the discharges from the RR SMR.

The design of the RR SMR is progressing, and a definitive source term will be available for Revision 2 of the Generic Environment Report. The preliminary discharge calculations presented in this report, including the underlying assumptions, parameter values, operating experience data and engineering design are subject to change. **{REDACTED FOR PUBLICATION}**.

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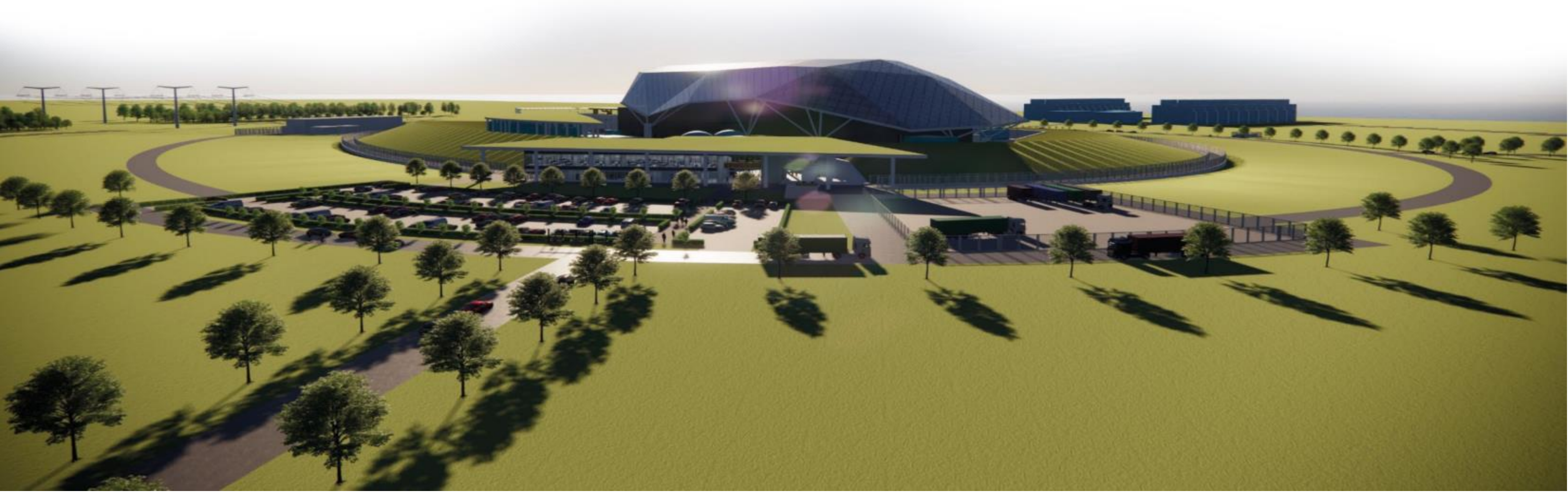

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29.1.1 Introduction

29.1.1 Introduction to Chapter

This report comprises Chapter 29 of the Rolls-Royce Small Modular Reactor (RR SMR) Environment, Safety, Security & Safeguards (E3S) Case. It forms part of the Generic Environment Report (GER) for the RR SMR and is a Tier 1 report in the E3S Case as defined in Chapter 1.

The report presents an overview of information on anticipated discharges of aqueous and gaseous radioactive waste to the environment from the normal operation of the RR SMR. It is based on design information available following completion of the Preliminary Concept Definition (PCD) in 2022 and will be updated using the information that will become available as the design matures.

29.1.2 Objectives

The information presented in this Chapter provides quantitative estimates of, and proposed limits for, aqueous and gaseous radioactive wastes, and compares the estimated discharges with those from similar power stations across the world. This Chapter supports the following E3S key objectives:

Table 29.1.1 -1. Key E3S Objectives relevant to Chapter 29

Objective ID	Category	Key Driver	Metric
4	Safety	Minimise normal operation and maintenance dose to ALARP Levels	A measure of impact on potential worker and public dose, taking account of time, distance and shielding.
5	Environment	Minimise environmental impact: showing compliance with Best Available Techniques (BAT) principles for minimisation of conventional and radioactive waste and ensuring sustainable development.	Estimates of the volume and activity of waste generated, and effluent discharged via different routes over plant lifecycle including decommissioning. Clear adherence to BAT principles.

The specific objectives of this Chapter are to:

1. Define and describe the normal operations source term (defined in Section 29.3.1) associated with the operation of the RR SMR.

2. Provide estimates of the aqueous and gaseous discharges to the environment, taking into account losses through treatment/abatement and radioactive decay.

3. Present proposed permit limits for discharge of aqueous and gaseous radioactive effluent to the environment, based on discharges from (2) above and justification of any provision within those limits to allow for variability and uncertainty in the discharges.

4. Compare the estimated radioactive discharges from the RR-SMR with those from similar nuclear power stations worldwide.

5. Summarise work which will be undertaken to refine the interim estimates as the design of the RR SMR progresses.

This first draft of Chapter 29 provides an outline of progress made towards these objectives based on the design of the RR SMR at PCD **{REDACTED FOR PUBLICATION}**. The report outlines the methodologies used, applying conservative assumptions where data are not yet available. As the design of the RR SMR progresses in line with the E3S-informed approach outlined above, it is anticipated that the estimates of discharges will reduce as the design develops.

29.1.3 Scope

This Chapter quantifies radioactive effluent discharges from all modes of normal operation, during the operational lifetime of the RR SMR. It excludes discharges arising from construction works or from decommissioning at the end of the operational life, which would be site-dependent and thus assessed separately.

Discharge of radioactive effluents takes into account the changes in radionuclide concentration and distribution during power operation and shutdown phases, which are considered to bound the different phases of operation.

The radioactive effluent discharge calculations include a contribution from unplanned but reasonably foreseeable operational occurrences (expected events), discussed further in Section 29.4. However, it does not include radioactive effluent arising from abnormal events, such as discharges arising from accident conditions.

Quantification of radioactivity embedded in structures and present in solid waste is outside the scope of this Chapter and is addressed in Chapters 11 (Radioactive Waste), 12 (Radiation Protection) and 20 (Chemistry).

Quantification of solid radioactive waste and spent fuel is not in scope of this chapter and will be addressed by an Integrated Waste Strategy (IWS), summarised in Chapter 26 (Radioactive Waste Management Arrangements).

Physical Boundary

SSCs from Reactor Island (R01) and Turbine Island (T01) have the greatest bearing on discharges to the environment and are therefore the focus of this document. Descriptions of the key SSCs relevant to quantification of radioactive discharges are summarised in this Chapter and considered in detail in Chapter 25.

29.1.4 Key Interfaces with Other Chapters

This chapter draws upon the information presented in other Chapters of the RR SMR E3S Case. Notable topic areas within E3S Case that support the production of this Chapter or are otherwise related to the Chapter are detailed in Table 29.1.1-2 below.

Table 29.1.1-2 Interfaces with other key Chapters

Chapter	Title	Summary of contents
1	Introduction and General Considerations	Provides an overall introduction to the suite of 33 chapters that make up the E3S Case.
4	Reactor Systems (Fuel & Core)	Presents the overarching summary and entry point to the design information for the fuel and core design of the RR SMR. Describes composition and configuration of fuel, control rods, and associated operational parameters, and will be a key supporting document for quantification of radioactive discharges.
9A	Auxiliary Systems	Presents the overarching summary and entry point to the design and safety information for the Auxiliary Systems of the RR SMR. Fuel handling and storage and ventilation are key systems supporting quantification of radioactive discharges.
11	Management of Radioactive Waste	Presents overall summary for all phases (gaseous, liquid, solid) RR SMR radioactive waste treatment, collection and drainage systems for radioactive wastes. Design and performance of these systems will be important in determining waste form and concentrations discharged to the environment.
15	Safety Analysis	Covers both deterministic and probabilistic safety assessment, as well as assessment of internal hazards and external hazards. Key elements of the chapter (principally the Fault Schedule) relevant to determining expected events which may contribute to radioactive waste production and discharge.
20	Chemistry	Presents the safety claims and arguments associated with the chemistry of the RR SMR, including how the chemistry regime minimises risks So Far As Is Reasonably Practicable (SFAIRP) and As Low As Reasonably Practicable (ALARP) using Best Available Techniques (BAT). Appropriate management of reactor chemistry ensures that the production of radioactive material and associated worker-dose and discharges can be minimised.
25	Detailed information about the design	Provides a technical description of the facility's main plants, systems and processes, which have a bearing on radioactive waste (solid, liquid and gaseous) generation, treatment, measurement, assessment and disposal

26	Detailed description of radioactive waste management arrangements (RWMA)	Describes RWMA for RR SMR. This Chapter describes the sources and predicted arisings of radioactive waste and provides an overview of application of radioactive waste minimisation and optimising disposals routes, including discharges to the environment. It will include quantification estimates for radioactive waste streams (solid waste, spent fuel, combustible waste etc) not included in this chapter.
27	Approach for Optimisation through the Application of BAT	Chapter provides detail on the Optimisation and BAT methodology used during the design of the RR SMR. Details claims, arguments, and evidence for RR SMR to demonstrate that BAT has been applied.
28	Sampling arrangements, techniques and systems for measuring and assessing discharges and disposals of radioactive waste	Presents information on RR SMR sampling & monitoring arrangements for in-process (key point upstream of the systems that interface with environment e.g., gaseous waste system) and final discharge monitoring (all forms to environment, independent monitoring), relevant to SSC design and monitoring/validation of environmental discharges.
30	Prospective radiological assessment	Presents the radiological assessment of doses to public and non-human species. Describes data and methods used to calculate doses to the public and environment from discharges of radioactive effluent and justifies why the model's data and assumptions used are appropriate. Quantification of effluent discharges from the RR SMR is a key input into these calculations.

29.1.5 Key Interfaces with Other Chapters

The information presented in this Chapter provides some of the evidence supporting the Fundamental Claims for quantification of discharges and prospective permitting limits. This Chapter summarises the work done to address the claims to date and highlights additional actions to be undertaken as the design progresses. Further detail on the Fundamental Claims for environmental aspects of the E3S Case can be found in Chapter 27 on BAT.

The Fundamental Claims for Chapter 29 are:

1. Quantitative estimates of discharges and disposals of aqueous and gaseous effluents are defined, bounded by the RR SMR normal operations source term.

2. Discharges of aqueous and gaseous radioactive effluent to the environment from the RR SMR are reduced to levels that are ALARA.

3. Radioactive effluent discharges from the RR SMR, corrected for differences in design, are comparable to discharges from similar nuclear power stations globally.

29.1.6 GDA Context

The following requirements in Table 29.1.1-3 are from the Environment Agency GDA Guidance for Requesting Parties [1] relating to quantification. This Chapter will, on final issue, meet the requirements laid out in Table 29.1.1-3 for aqueous and gaseous wastes. The Table also highlights where information for other wastes not covered in this chapter can be located.

Table 29.1.1-3. GDA Requirements for quantification of radioactive waste disposals

GDA Requirement	Information Source
Quantitative estimates of waste arisings for normal operation are required, including: • *discharges of gaseous and aqueous radioactive wastes* • *arisings of combustible waste and disposals by on-site or off-site incineration* • *arisings of other radioactive wastes - by category and disposal route (if any) - and spent fuel* *The estimates must include:* • *Estimates of operational fluctuations, trends and events that occur over the lifetime of the facility, including start-up, shutdown, maintenance and expected events* • *The approach taken to identify these fluctuations and assessment of their impact on discharges and waste.* • *Via each discharge point and discharge route*	Chapter 29, Sections 29.3 and 29.4 (gaseous & aqueous radioactive wastes) Chapter 26 (solid radioactive waste & other radioactive waste for disposal at permitted facility)
For gaseous and aqueous radioactive waste, the RP must estimate the monthly discharges: • *On an individual radionuclide basis for significant radionuclides*[1] • *On a group basis (for example 'total alpha' or 'total beta') for other radionuclides* • *Via each discharge point and discharge route*	Chapter 29, Section 5
The radionuclide selection should be consistent with 2004/2/Euratom [2]. *The RP must quantify the activity of important individual radionuclides and overall groupings of radionuclides (for example, total beta), together with mass and/or volume.*	Chapter 29, Section 29.5
For combustible and other radioactive wastes, the RP must estimate the annual arisings and disposals during operation and give an indication of the likely arisings during decommissioning.	Chapter 26

[1] The definition of 'significant radionuclides' is given in Section 5.1 of this Chapter.

GDA Requirement	Information Source
The RP must identify wastes in terms of their: • *category - High Level Waste, Intermediate Level Waste, Low Level Waste, Very Low Level Waste* • *physico-chemical characteristics* • *proposed management and disposal route*	Chapter 26
• *Estimates of discharges and disposals should clearly show the contribution of each constituent aspect of normal operations, including:* • *Routine operations (typically, the design basis or 'flowsheet design' and the minimum level of disposals)* • *Start-up and shutdown* • *Maintenance and testing* • *Infrequent but necessary aspects of operation for example plant start-up, trips, maintenance, shutdown and refuelling* • *Foreseeable (based on a fault analysis) unplanned events during normal operation that remain consistent with using BAT for example occasional fuel pin or plant failures*	Chapter 29, Sections 29.3 & 29.4
The RP must support estimates with performance data from similar facilities, where such facilities exist. They must also explain, where relevant, how changes in design or operation from those facilities affect the expected discharges and disposals.	Chapter 29, Section 29.5
The RP must demonstrate that discharges and waste arisings will not exceed those of comparable power stations across the world [3].	Chapter 29, Section 29.7 (gaseous and aqueous radioactive effluents) Chapter 26 (solid and other radioactive waste for offsite disposal)
The RP must provide proposed limits for: • *Gaseous discharge* • *Aqueous discharge* • *Disposal of combustible waste by onsite incineration, if proposed*	Chapter 29, Section 29.6 (gaseous and aqueous discharges) Chapter 26 (combustible waste)
The RP must provide proposals for annual site limits (on a rolling 12-month basis) for gaseous and aqueous discharges. The RP must describe how they derived these limits. They can	Chapter 29, Section 29.6

GDA Requirement	Information Source
also propose limits to reflect an operating cycle (campaign limits).	

Currently the design maturity of the RR SMR is at Preliminary Concept Design (PCD) and the information presented in this chapter reflects this maturity.

29.2 Regulatory Context

29.2.1 Overview of Regulatory Context

The relevant international and national obligations, legislation and policy decisions to protect people and the environment from harm resulting from radioactive discharges are summarised below.

International context

The International Commission on Radiation Protection (ICRP) Basic Safety Standards (BSS) [4] set out the principles and fundamentals for radiation protection, which are adopted worldwide. The EC Directive 2013/59/EURATOM [5] applies these principles to a system of radiation protection and control, to be transposed into the domestic legislation of the European Union Member States'[2] . Of particular relevance to this Chapter are the requirements for States to keep doses to members of the public as low as reasonably achievable (ALARA) through optimisation of practices involving radioactive substances, and to ensure protection from unacceptable risks of exposure to radioactivity by enforcing limits. The BSS now also explicitly extends protections from exposure to 'the environment', that is, species other than human beings.

The UK is also signatory to Convention for the Protection of the Marine Environment (OSPAR Convention), on measures to protect and improve the environment of the North-East Atlantic. Amongst other obligations, OSPAR requires Contracting Parties commit to prevent and eliminate pollution of the North-East Atlantic from land-based sources, including radioactive discharges from nuclear power stations through the use of Best Available Techniques (BAT) and Best Environmental Practice.

Regulation in England and Wales

Discharges of radioactive substances from nuclear power stations to the environment are controlled under the Radioactive Substances Regulations (RSR), set out in Schedule 23 of the Environmental Permitting (England and Wales) Regulations 2016 (as amended) (EPR16). The regulations ensure operators do not produce unnecessary radioactive waste, and that any waste generated is minimised and managed safely. Where radioactive waste is discharged to the environment, operators must demonstrate that BAT has been used to manage that waste, and to ensure that the impact of discharges on members of the public or non-human species have been minimised to levels that are ALARP.

EPR16 requires Regulators to, in the exercise of their function, ensure that:

1. All exposures to ionising radiation of any member of the public and of the population as a whole resulting from the disposal of radioactive waste are kept as low as reasonably achievable, taking into account economic and social factors;

[2] Transposition of 2013/59/Euratom into UK legislation was completed by the deadline of February 2018, prior to the coming into force of the European Union (Withdrawal Agreement) Act 2020 on 31 December 2020

2. The sum of the doses arising from such exposures does not exceed the individual public dose limit of 1mSv per year;

3. The dose to an individual due to discharges from any source (since 13th May 2000) does not exceed 0.3mSv per year

4. The dose to an individual due to discharges from any single site does not exceed 0.5mSv per year.

To demonstrate these conditions are met, the Environment Agencies limit the types, physical form and quantities of radioactivity which may be discharged to the environment in a given period.

In 2009, the then Department for Energy and Climate Change (DECC)[3] issued Statutory Guidance to the Environment Agency[4] concerning the regulation of radioactive discharges to the environment [3], which provides that

"the Environment Agency should set discharge limits based on the use of BAT by holders of authorisations under the Radioactive Substances Act 1993 [superseded by EPR16]" … and that "*Limits should be set at the minimum levels necessary to permit "normal" operation or decommissioning of a facility".*

The Statutory Guidance then provides a lower bound of exposure for the most exposed members of the public of 10μSv per year, below which the Environment Agency should not seek to further reduce the discharge limits that are in place, provided that the holder of the permit continues to apply Best Available Techniques (BAT).

The statutory guidance issued by DECC [3] further requires that:

"The application of BAT in England and Wales will ensure that discharges from new nuclear power stations constructed in the UK will not exceed those from comparable power stations across the world."

This requirement ensures the UK's continued adherence to OSPAR requirements for a progressive reduction in radioactive discharges, whilst not compromising UK domestic energy policy[5], and is captured in the GDA requirements.

29.2.2 Alignment with the Regulatory Objectives and Principles

The Environment Agency RSR Objective and Principles (ROPs) [6], set out the regulatory principles the Environment Agency applies in the delivery of their function as laid out in EPR16 and government policy.

The ROPs are supported by a set of RSR Generic Developed Principles (GDPs), which set out the Environment Agency's expectations on permit holders carrying out radioactive substances

[3] Subsumed into the Department for Business, Energy & Industrial Strategy (BEIS) in July 2016.
[4] Since 2013, regulation of radioactive substances in Wales has been the responsibility of Natural Environment Wales
[5] Scottish Government energy policy does not allow for new nuclear build and thus the policy for discharges from new build in England and Wales can be taken to represent total new build discharges in UK.

activities [7]. Reference [8] indicates how the RR SMR Generic Environment Case (GER) will incorporate the GDPs throughout the document identifying the key principles for each Chapter of the GER. The key GDPs directly relevant to Chapter 29 on the quantification of discharges to the environment are identified in Table 29.2-1.

Table 29.2-1. RSR Generic Developed Principles relevant to GDA Requirements on quantification of radioactive effluent discharges

GDP ID	Title	Principle
RSMDP3	Use of BAT to Minimise Waste	*BAT should be used to ensure that production of radioactive waste is prevented and where that is not practicable minimised with regard to activity and quantity.*
RSMDP12	Limits and Levels on Discharges	*Limits and levels should be established on the quantities of radioactivity that can be discharged into the environment where these are necessary to secure proper protection of human health and the environment.*
RPDP1	Optimisation of Protection	*All exposures to ionising radiation of any member of the public and of the population as a whole shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account*
ENDP10	Quantification of Discharges	*Facilities should be designed and equipped so the best available techniques are used to quantify the gaseous and liquid discharges produced by each major source on a site.*

The ROPs and GDP's in particular those listed above will be taken into account in meeting the objectives for quantification of discharges to the environment in this Chapter.

29.3 Development of a Source Term for Estimating Radioactive Discharges

29.3.1 Source term overview

Source term definition

"Source term" refers to the types, quantities and physical and chemical forms of radionuclides present in a nuclear plant that have the potential to give rise to exposure to ionising radiation, as well as the generation of radioactive waste and discharge of radioactive effluents to the environment [9]. The normal operation source term covers all forms of radioactivity present within the primary circuit and associated plant systems during normal operations, including radioactivity which is:

1. present in the coolant and entrained gases
2. fixed or deposited onto surfaces of structures such as reactor core and primary circuit pipework
3. accumulated in abatement structures such as filters and demineralisers.

Different types and categories of normal operation source term have been established, often denoting the origin or intended application. The types and categories of the normal operation source term associated with the RR SMR are summarised below.

Source term types

The RR SMR normal operations source term will be derived logically, focussing on the source term in the reactor core and using this as a basis for deriving the source terms for supporting plant systems. As such, the overall source term for the RR SMR will be made-up of the following constituent source terms [9]:

1. **Primary Source Term (PST)**: The PST considers the initial formation of radionuclides in the primary coolant in the reactor core. It includes fission products, corrosion products, activation products and actinides. The PST represents the most important dataset in the normal operation source term suite and is used as the starting point for all other system source terms subsequently derived, as summarised below.
2. **Primary System Source Term (PSST)**: The PSST covers the radioactive inventory in the coolant and gaseous streams in primary circuit systems, as well as radioactivity deposited on the inner surface of system components which becomes fixed on primary circuit surfaces.
3. **Secondary System Source Term (SSST)**: The SSST covers the radioactive inventory in the coolant and steam of secondary circuit systems which may be produced in the event of a steam generator (SG) tube leak.
4. **Fuel Crud Source Term (FCST)**: The FCST covers the radioactive inventory of the fuel crud (material deposited on the surface of the fuel), predominately made up of corrosion products (CPs). Fuel deposits impact the PST due to the activation and resuspension of deposited material from the fuel surface. This behaviour will be reflected in the PST.

5. **{REDACTED FOR PUBLICATION}**

The Interim PST, described in Chapter 20 is used throughout this Chapter as the basis for the quantification of radioactive discharges to the environment. The Interim PST will be refined as the design progresses to produce a definitive PST.

Categories of Source Term Value

One of the primary functions of the source term is to facilitate technical assessments across several disciplines to support the development of the RR SMR and demonstrate compliance with legislative requirements. These assessments require different expressions of the source term and, as consequence, the RR SMR normal operation source term values are categorised into the following groupings:

1. **Best Estimate (BE)**: This will give an overall best estimate of the source term expected in the RR SMR. This will be a representative condition that is realistic so as not to result in over-specification of the source term for plant systems.

2. **Design Basis (DB):** This will give a conservative maximum value for the source term which can be considered to be a bounding limit for the plant design. The DB value is not expected to be exceeded during operation, even during transients and when expected events such as fuel failures occur. The DB value is important for key safety related applications such as shielding calculations to ensure that doses to the operators and public are minimised.

3. **Cycle Average (CA)**: This will give a source term that is averaged over the entire fuel cycle, including start-up, power operation, shutdown and refuelling outage phases. The CA values will be determined for both BE and DB values and will include expected events such as fuel failures and unplanned shutdowns.

The quantification of radioactive effluent discharge to the environment presented later in this document will be based on the BE source term, in line with relevant good practice (RGP).

29.3.2 Radionuclide production mechanism

Radionuclides present in the primary coolant of PWRs may be categorised into 4 distinct groups based on their production mechanisms:

1. Fission products (FP)
2. Actinide products (ActP)
3. Corrosion products (CP)
4. Activation products (AP).

An overview of the generation of these radionuclides in the reactor is summarised below.

Fission products

Fission products originate from the fission of fissile materials in the reactor core. FPs present in the primary coolant are generated via two principal mechanisms:

1. Fissioning of trace amounts of uranium that may be present on the external surfaces of the fuel assemblies, so called "tramp uranium" or uranium impurities in the fuel cladding material, and

2. Leakage of volatile fission products (generated during core irradiation) though small pinhole defects in the fuel cladding or following fuel failures.

Typical FPs in PWRs include radioisotopes of krypton, xenon, iodine, caesium and strontium.

Actinide products

Actinide products (ActP) originate from the neutron activation of fertile radionuclides (primarily uranium-238) present in the fuel material, as impurities in the fuel cladding material or as tramp uranium on a fuel assembly surface. Further ActP are generated by bombardment of daughter products (neptunium and plutonium) by neutrons and alpha particles in the reactor core. Examples of typical ActP in PWRs include radioisotopes of americium, curium, neptunium and plutonium.

Corrosion products

Corrosion products are formed when metallic impurities are released into the primary coolant as a result of surface degradation (e.g., corrosion and wear & tear) of primary circuit materials. These impurities are carried by the primary coolant through the intense neutron field present in the reactor core resulting in the activation of composite metals such as cobalt and iron. A proportion of these impurities will deposit onto the surfaces of the reactor core and other primary circuit structures, while the remainder will be retained in the coolant as suspended or solubilised material. Examples of typical CPs include radioisotopes of iron, cobalt, chromium and manganese.

Activation products

Activation products (AP) are produced from neutron activation of the primary coolant itself and impurities entrained in the coolant. The impurities may include material present in the coolant incidentally, or substances intentionally added to control coolant chemistry and to minimise the degradation of structural material. Some of the key AP in discharges to environment from PWRs include:

1. Tritium (H-3), which is mainly produced by neutron activation of boron and lithium normally added to PWR cooling water. However, these mechanisms are not significant sources of tritium in the RR SMR, which has adopted a boron-free, potassium-based coolant chemistry. Thus, the most important tritium production mechanism for the RR SMR is neutron activation of naturally occurring deuterium (H-2) in the primary coolant water **[H-2 (n, γ) -> H-3]**. Another important production mechanism is as a by-product of ternary fission in the fuel; however, the bulk of tritium thus produced is retained within the fuel pin and only a small fraction diffuses through the zirconium alloy fuel cladding into coolant during normal operations.

2. Carbon-14 (C-14), the bulk of which is produced by the neutron activation of oxygen-17 naturally present in the primary coolant water **[O-17 (n, α) -> C-14]**. Other production mechanisms include neutron activation of nitrogen-14 and carbon-13 impurities in primary coolant water.

3. Argon-41 (Ar-41) is produced by the neutron activation of naturally occurring argon (Ar-40) dissolved in primary coolant water or present in the air around the reactor core **[Ar-40 (n, g) -> Ar-41].** The main source of Ar-41 is considered to be the activation of Ar-40 in the air around the Reactor Pressure Vessel (RPV). De-aeration and degassing of the primary coolant are expected to limit the quantities of dissolved gases in the coolant.

Radionuclide groupings

It is recognised that some radionuclides may be generated via more than one production mechanism. Some important examples of this include radionuclides such as zirconium-95 (Zr-95), which is both a fission product and a corrosion product, and tritium which is both a fission and an activation product. Such radionuclides have been placed in the radionuclide group associated with the dominant production mechanism for simplicity, with the production from both mechanisms reflected in the source term values [11].

29.3.3 Interim PST

An initial, or interim PST has been developed to support the design of RR SMR systems' structures and components (SSCs) (notably waste management systems), and to facilitate the technical assessments required to support early stages of the GDA.

{REDACTED FOR PUBLICATION}

{REDACTED FOR PUBLICATION}

The Interim Source Term has been developed in parallel with the work done to identify the radionuclides for inclusion in the definitive RR SMR normal operation source term. The list of radionuclides will be kept under review to ensure that it remains representative of the RR SMR design as it evolves. Further details on source term development can be found in Chapter 20.

{REDACTED FOR PUBLICATION}

Radionuclide selection for normal operation source term

A list of radionuclides that form the basis for the normal operation source term have been derived using a consistent and logical methodology based upon the key steps outlined below:

1. Relevant codes and standards on normal operation source terms are reviewed and radionuclides included in these codes and standards are selected to form an initial SMR radionuclide list.

2. Source term datasets from modern nuclear power station designs of similar technology are then reviewed and radionuclides in their source terms are added to the initial SMR radionuclide list.

3. Consideration of the design of the RR SMR is made and any differences in the design relative to those reviewed in the second key step are analysed with respect to the source term and radionuclide generation. Radionuclides that are produced as a consequence of any differences between the RR SMR and the nuclear power stations in the source term dataset,

and which are not included in either of the sources above, are added to the initial radionuclide list.

{REDACTED FOR PUBLICATION} Detailed description of the method and the resulting output are provided in Reference [11]. The list of radionuclides will be kept under review to ensure that it remains representative of the RR SMR design as it evolves.

{REDACTED FOR PUBLICATION}

{REDACTED FOR PUBLICATION Table 29.3-1. Interim RR SMR PST}

29.3.4 Planned development of the definitive normal operations source term

It is noted that not all information required to develop the normal operations source term is currently available, and the calculations and analyses are dependent on available PWR OPEX data and assumptions based on early-stage design. This section describes the process for determining the normal operations source term and the additional work or data collection required to produce the definitive version.

A definitive RR SMR-specific PST is currently in development. The definitive PST will underpin future analyses and assessments carried out to support Revision 2 of the RR SMR GER. **{REDACTED FOR PUBLICATION}**

Source term derivation

The RR SMR PST will be developed using a combination of approaches. OPEX data, is used where available, with modelling and experimental data used where OPEX information is incomplete or not directly applicable. In order of preference, the approaches selected are:

1. Direct OPEX from analogous plants (directly applicable to RR SMR with minimal adjustment e.g., scaling with thermal power)

2. Indirect OPEX from analogous plants (applicable but requires further treatment to derive the activity for a radionuclide e.g., use of chemical analogues for unmeasured radionuclides)

3. Computer codes/models (empirical, based on plant or experimental data)

4. Computer codes/models (theoretical)

5. Calculation from first principles

{REDACTED FOR PUBLICATION}

Source term reduction/optimisation philosophy

The development of the RR SMR from fundamental PWR technology has afforded the opportunity to identify and implement relevant good practice and source term reduction measures into the fundamental design of the RR SMR This ensures disposals are ALARP and supports the optimisation of radiation exposure of members of the public and the environment.

The generation of FP and ActP is an unavoidable consequence of nuclear power generation; however, the impact of these products can be minimised through improved fuel fabrication processes, careful design of reactor systems, and optimised coolant chemistry and operating regime. RR SMR source term optimisation therefore focuses on the minimisation of CP and AP which primarily arise from the activation of impurities deliberately or unintentionally added to, and often unavoidably present in, the reactor coolant.

Complete elimination of most radionuclides that will be generated during the operation of the RR SMR is either not feasible or will be disproportionately costly. Thus, the presence of radionuclides in the reactor coolant is accepted but must be managed in accordance with the Rolls-Royce SMR requirements to minimise radiological risks to levels that are As Low As Reasonably Practicable (ALARP) and to use BAT [14]. A hierarchy of controls has therefore been conceived, the application of which will, and can be demonstrated to ensure optimisation of the source term in-line with the RR SMR Environment, Safety, Security and Safeguards (E3S) principles [15], as illustrated in Figure 29.3-1. The controls adopted are:

1. Elimination
2. Prohibition and Substitution
3. Localisation and Minimisation
4. Immobilisation; and
5. Mitigation.

The above controls are integrated into the design philosophy of the RR SMR and are underpinned by a set of mandatory requirements and good practice recommendations aimed at driving source term reduction and optimisation in-line with the RR SMR E3S principles [14]. These requirements apply to all systems connected to the Reactor Coolant System (RCS) which contains the reactor coolant, as well as systems that can potentially discharge their contents into the RCS (either by design or due to fault conditions) and influence the composition of reactor coolant. They are to be used in concert within an overall philosophy of applying BAT to minimise the source term to levels that are As Low As Reasonably Achievable (ALARA). Details of the mandatory requirements and recommendations are presented in Reference [15] . Details of the application of these principles for each SSC are recorded in DOORs

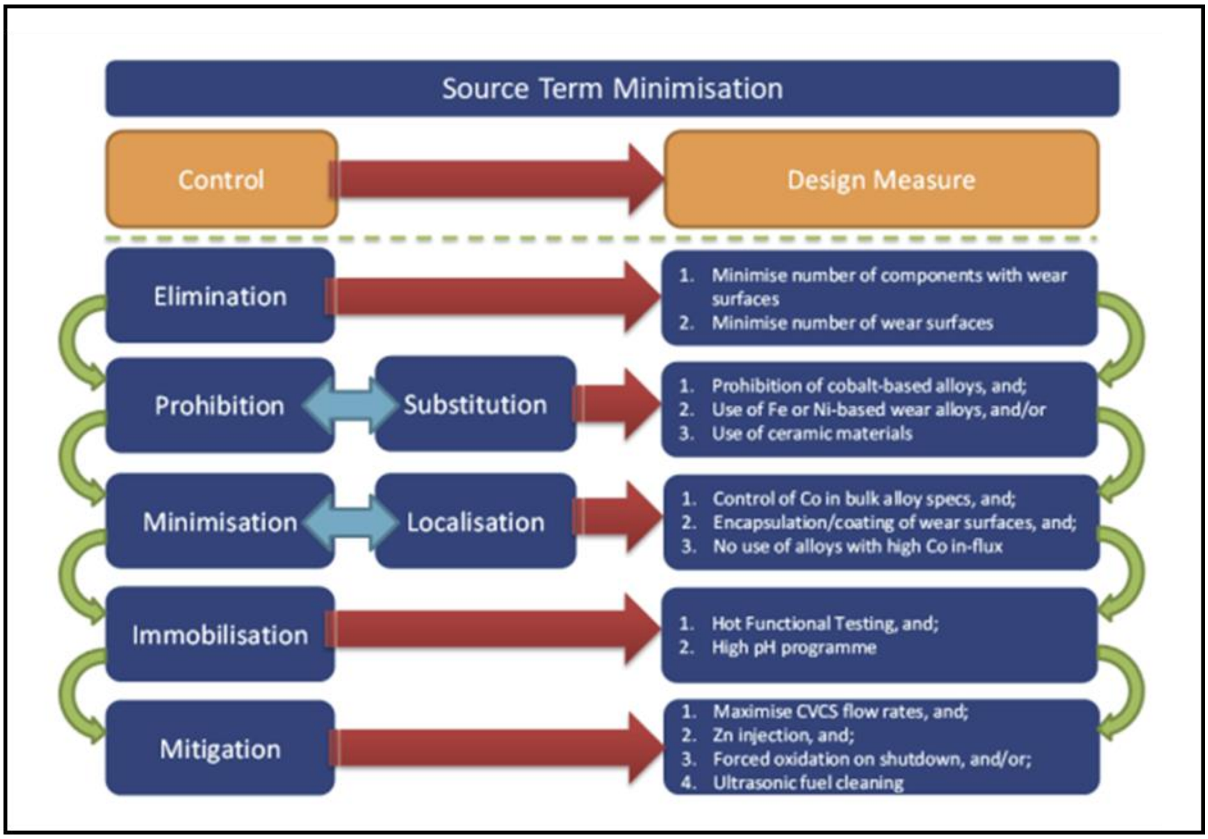

Figure 29.3-1. An illustration of the application of the hierarchy of source term controls for Co-60

The definitive PST will account for the opportunities for reduction and minimisation of the source term through plant design, chemistry, choice of fuel etc.

Operating phases

Six operating modes, characterised by RCS temperature, pressure and refuelling status, are defined in the RR SMR Reactor Island Operating Philosophy [16]. The available OPEX does not allow derivation of source term for each mode, thus, for the purposes of defining the normal operation source term, these six modes have been condensed into the following four operating phases covering the full fuel cycle [9]:

1. Start-up
2. Power operation
3. Shutdown
4. Refuelling outage

For a standard cycle, the start-up phase covers the period between the completion of the refuelling outage and steady-state power operation at 100% power. Start-up is initiated once the core has been loaded with fresh fuel, the RPV head secured, and the plant filled such that it is water solid. Once this state has been attained, warm-up operations can commence such that the reactor coolant transitions from a temperature of ≤ **{REDACTED FOR PUBLICATION}**

°C and **{REDACTED FOR PUBLICATION}** pressure up to those associated with the power operation phase.

The power operation phase covers steady-state power operation at 100% power with a normal operating pressure of **{REDACTED FOR PUBLICATION}** MPa and an average reactor coolant temperature of approximately **{REDACTED FOR PUBLICATION}** °C. The power operation phase is the phase of longest duration in the cycle of the RR SMR.

The shutdown phase covers the period between steady-state power operation and the subsequent refuelling outage. Shutdown is initiated when the control rods are fully inserted and the temperature and pressure transition from those associated with the power operation phase down to approximately **{REDACTED FOR PUBLICATION}** °C and **{REDACTED FOR PUBLICATION}** pressure. Additionally, the shutdown phase covers partial drainage of the plant to the required level in preparation for RPV head removal during the outage phase.

The refuelling outage phase covers outage activities such as RPV head removal, flood up, removal of spent fuel, relocation of used fuel and the loading of fresh fuel. **{REDACTED FOR PUBLICATION}**.

Determination of the definitive PST will take into account the changes in radionuclide concentration and distribution across the RR SMR during the different phases of operation and will take into account any differences between the current conservative estimates and RR SMR-specific information which becomes available as the design develops.

Expected events

Additional contribution from events reasonably expected to occur at least once during the operating life of the plant (occurrences with an initiating event frequency of 1E-02 or more per reactor year) must be accounted for in the derivation of the primary source term. These 'expected events' are akin to anticipated operational occurrences described in the Safety Analysis topic area.

A shortlist of credible expected events for the RR SMR Environment Case was derived from published literature, the outcome of an RR SMR stakeholder workshop held in November 2020 and information contained in design documentation produced by the RR SMR safety and chemistry teams [18]. The main expected events identified as being relevant to the RR SMR normal operation source term (based on the screening criterion of 1E-02 per reactor year) are:

1. Fuel pin failure. Whilst advances in fuel and cladding design have significantly reduced the incidence of fuel failure, this event cannot be claimed to have been eliminated and multiple fuel pin failures could occur over the 60-year life of the RR SMR.

2. Primary-to-secondary circuit steam generator tube leak. This event is expected to result in the transfer of radioactivity from the RR SMR primary to secondary circuit. A Secondary Systems Source Term (SSST) covering the radioactive inventory in the coolant and steam of secondary circuit systems (including gaseous volumes) will be derived through application of a primary-to-secondary leak rate factor.

3. Unscheduled/automatic reactor shutdown. The number of unscheduled shutdown events that meet the expected events criterion over a refuelling cycle will be determined from the Fault Schedule and the associated event frequencies summated. An appropriate correction factor based on the summated event frequencies will be calculated, using either relevant

OPEX or appropriate modelling, and applied to the BE shutdown source terms for normal operations.

A fourth potential expected event associated with the leakage of boron and/or its activation products (lithium-7 and tritium) into the reactor coolant from defective boron control rods [19] has been identified. The extent to which this phenomenon is a credible expected event depends on factors such as the exposure conditions within the RR SMR core and the operation for which boron control rods are deployed (e.g., shutdown or reactivity control, or both). The credibility of defective boron control rods as an expected event for the RR SMR design will be re-evaluated once the RR SMR Fuel and Core design (including supporting analyses) is better defined.

The list of RR SMR expected events shall be kept under review and periodically updated using information captured in the Fault Schedule. **{REDACTED FOR PUBLICATION}**

The approach for estimating the contribution of expected events to the RR SMR Source Terms and predicted discharges of radioactive effluent to the environment will depend on the origin of the data underpinning the Source Term. Source Terms based on OPEX data will include contributions from all but the most infrequent applicable expected events. In contrast, Source Terms generated from models and theoretical calculations will require further adjustments (e.g., using correction factors) to account for expected events. Considering the diversity of approaches that will be used to generate the definitive Source Term (see preceding Section), it is likely that both approaches for estimating the contribution of expected events to radioactive discharges to the environment will be adopted.

29.4 Interim Quantification of Discharges to the Environment

29.4.1 Overview of calculational approach

The calculational approach adopted is based on the application of radioactive effluent transfer and removal factors (derived from empirical observations from operating PWRs and engineering judgements published in peer-reviewed reports) to the PST and design parameters of the RR SMR (effluent masses and flow rates) in mass and activity balance models. The overall approach is illustrated in Figure 29.4-1.

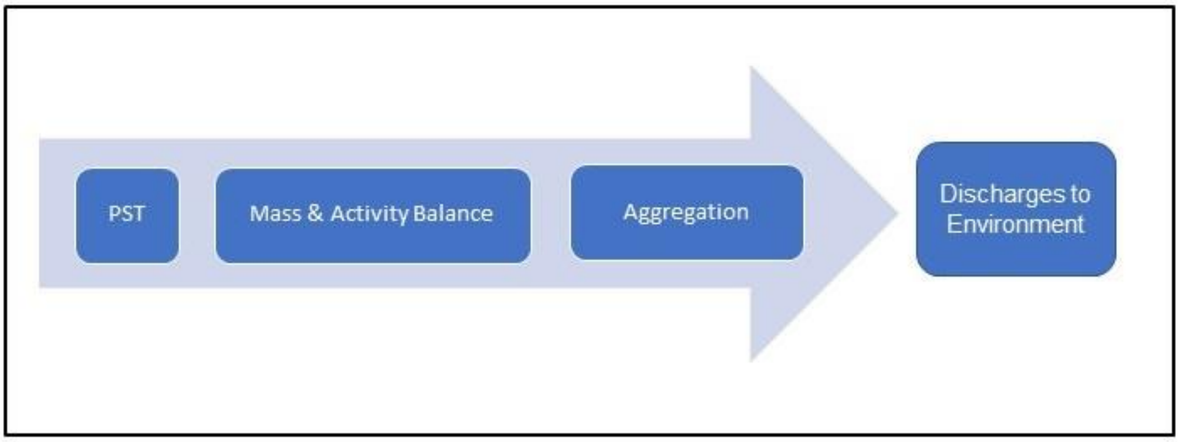

Figure 29.4-1 Overview of the approach for estimating effluent discharge to the environment

The interim PST presented in Table 29.3-1 forms the basis for the estimates of discharges to the environment described in this Chapter.

Discharges of radioactive effluent to the environment are estimated using a Mass and Activity Balance (MAB) model to calculate the activity of treated effluent streams exiting the aqueous and gaseous radioactive effluent treatment systems. This approach is based on the principle of conservation of mass and activity - the sum of mass and activity of the 'dirty' effluent streams entering treatment systems are equal to the mass and activity of the 'clean' output stream, plus the mass and activity retained by the treatment system.

The MAB functions through the application of Decontamination Factors (DFs) and Volume Reduction Factors (VRFs) to represent the partitioning of radioactivity from the aqueous and gaseous throughput to the solid/ semi-solid phase of the treatment system (activated charcoal beds, Ion eXchange (IX) resins beds, evaporator concentrates and Reverse Osmosis (RO) retentates and particulate filters).

DF and VRF used in this Chapter are typical values based on OPEX or taken from published literature, as outlined through this section.

Additional losses through radioactive decay are modelled where appropriate, for example in the case of the radioactive Gaseous Waste Treatment System (GWTS) where radioactive decay is the operating principle, and for radionuclides with short half-lives where decay is an important mechanism.

A final step in the calculational approach applies to gaseous discharges and involves the aggregation of the various treated gaseous effluent streams identified in Section 29.4.3 to derive the total discharges from the RR SMR to the environment. Treated aqueous effluent streams are transferred to the liquid effluent monitoring and discharge system (LMDS) [KNF30] tanks for storage and recycling, or discharge to the environment on rare occasions, and do not therefore require an aggregation step.

29.4.2 Sources and estimates of aqueous radioactive effluent discharges

Sources of aqueous effluent

Aqueous radioactive effluents are categorised and segregated based on their source and expected levels of contamination. Two primary sources of aqueous radioactive effluents have been characterised for the RR SMR [20]:

1. Primary liquid effluent originating from the Chemistry and Volume Control System (CVCS) [KB] letdown and the drainage of primary coolant to the reactor coolant drain tank (RCDT). The water quality in these systems is maintained to a high standard by the relevant purification and treatment system [KBE] and is therefore expected to have low levels of contamination.

2. Spent liquid effluent comprises a few effluent streams originating from different sources, collected for treatment by the sumps and vessels of the Collection and Drainage System (CDS) [KTA]. Spent liquid effluent comprises the following categories:

    a. Process drains – effluent derived from equipment drainage for maintenance, testing, or safety relief etc.; these effluents are expected to have low levels of contamination.

    b. Chemical drains – effluent derived from chemical sampling and equipment decontamination; these effluents are expected to have higher levels of contamination than process drains.

    c. Floor drains – effluent derived from leaks and floor washings. Floor drains are segregated into active and non-active, depending on layout (dependent on radiological contamination zoning). These effluents are expected to have varying levels of contamination.

Table 29.4-1 summarises the estimated volumes of primary and spent liquid effluents generated during the course of an 18-month operating cycle.

{REDACTED FOR PUBLICATION Table 29.4-1. Estimates of radioactive aqueous effluent arisings over an operating cycle}

Further details on the estimation of the aqueous radioactive effluent volumes at this stage of design are presented in Reference [21].

The Steam Generator (SG) feedwater or secondary coolant is normally non-radioactive and is routinely let-down and transferred to the Wastewater Treatment Plant (WWTP) [GM-] for treatment to maintain the desired Water Quality Specification (WQS). However, in the event of a SG tube leakage, a small fraction of the radioactivity present in the primary coolant will be transferred to SG feedwater and distributed across the secondary coolant circuit. The approach for handling contaminated SG BlowDown (SGBD) depends on the degree of leakage and level of radioactive contamination of the secondary coolant (or any set operating limits). SGBD that is contaminated from small tube leaks is transferred to the LMDS [KNF30] for discharge, whilst SG blowdown that is contaminated from tube rupture is transferred to KNF20 for treatment and then KNF30 for discharge.

A primary-to-secondary leakage rate of 34 litres per day is assumed based on published data [22]. This value is subject to review as the design of the RR SMR progresses.

Processing and Treatment of aqueous effluent

The effluents summarised in Table 29.4-1 are processed through the KNF aqueous waste treatment system. The KNF system collects the aqueous radioactive effluents in tanks and treats them with a combination of separation methods for removal of radionuclides and chemical contaminants. The treatment enables storage and recycling of effluents within RR SMR systems and processes, or otherwise renders the effluent suitable for discharge to the environment. The system manages both primary and spent liquid effluent and treats them based on their characterisation of the expected levels of radiological and chemical contamination [20].

The baseline architecture for KNF consists of storage tanks for collection of liquid effluent, with a combination of filtration, membrane separation, ion exchange, evaporation and degassing for treatment. Treated liquid effluents are recycled as make-up demineralised water in Reactor Island, or in specific cases discharged to the environment.

The KNF system is made up of the following sub-systems:

1. Processing & treatment system for primary liquid effluent (KNF10), which comprises:

    a. Storage tanks for primary liquid effluent, with transfer pumps and connection to sampling systems and KNF20 treatment systems.

    b. A vacuum degasser for removal of dissolved gases from effluent and make-up water.

2. Processing & treatment system for spent liquid effluent (KNF20), which comprises:

    a. Storage tanks for combined process and floor drains, with transfer pumps and connection to sampling systems.

    b. Storage tanks for chemical drains with one tank aligned to collect retentates from the RO process, with transfer pumps and connection to sampling systems.

    c. Abatement process equipment including backwashable filters, RO units, ion exchange demineralisers and vacuum evaporators (for reducing the volume of RO retentates).

3. Liquid effluent monitoring and discharge system (KNF30), which comprises:

    a. Active monitoring tanks for treated liquid effluent, with transfer pumps and connection to sampling systems.

    b. Discharge line to the cooling water island outfall, with sampling and monitoring.

    c. Recycle line to distribute as primary make up water, demineralised water in R01 and to CVCS [KB] or Spent Fuel Pool (SFP) systems.

The types of contaminants in effluents are identified as:

1. Radionuclides (fission, actinides, corrosion and activation products), including radioactive solutes, dissolved gases and suspended solids.

2. Chemical contaminants, including total dissolved solids (TDS), total suspended solids (TSS) and total organic carbon (TOC) content.

The key effluent routes are highlighted in the process flow diagram in Figure 29.4-2.

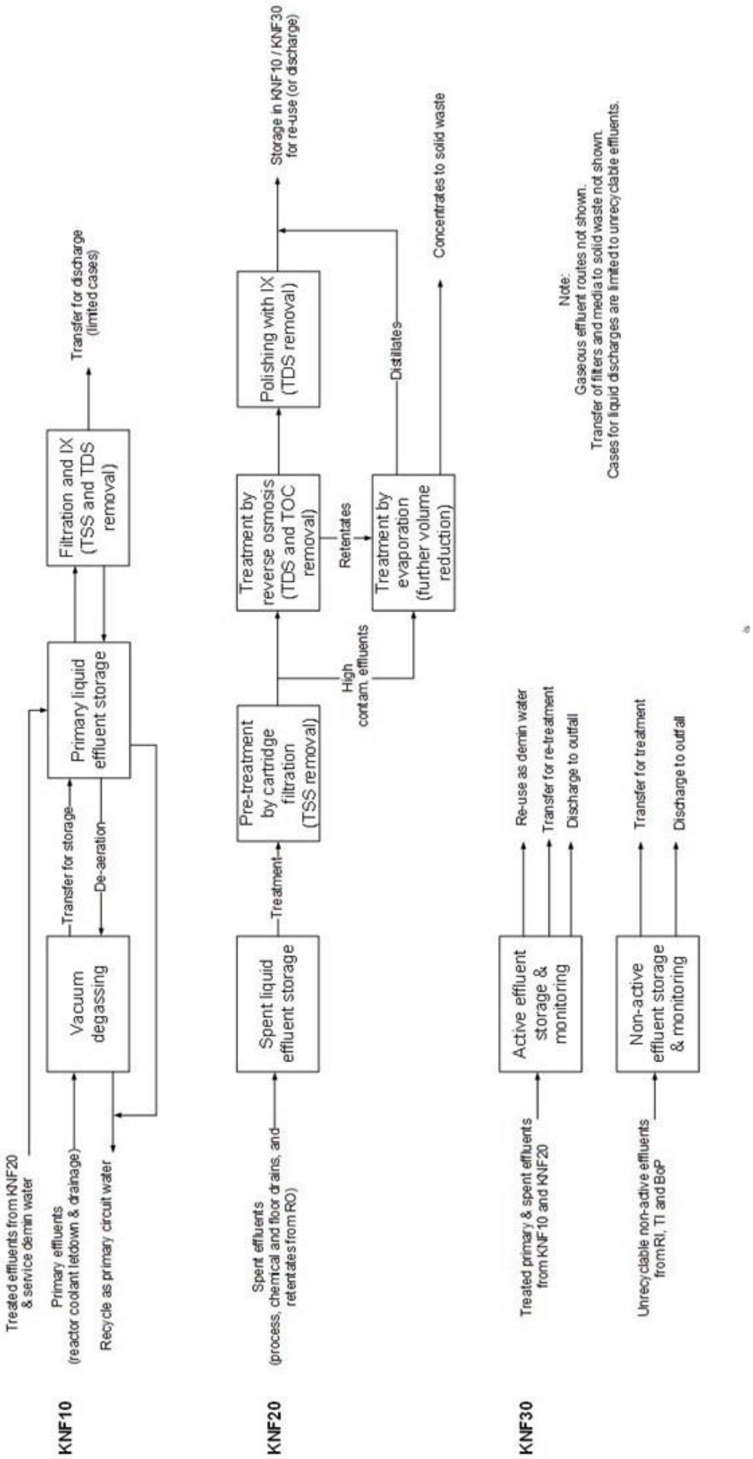

Figure 29.4-2 Process block flow diagram for aqueous effluent treatment systems [KNF]

Effluents treated in KNF20 are either reused in the primary circuit or diverted to the liquid effluent monitoring and discharge system KNF30 for storage and subsequent recycle as demineralised water elsewhere in the Reactor Island. This is in accordance with the corporate objective of designing a product that can be operated with minimal discharge of aqueous radioactive effluent to the environment.

The level of decontamination of aqueous throughput achieved by the abatement techniques in KNF20 is provided in Table 29.4-2. Noble gases and other non-condensable gases are assumed to fully partition (100%) to the gas phase and to be unaffected by the treatment techniques – with the exception of carbon-14 for which only a small proportion (< 10%) is assumed to remain in the dissolved phase [23]. Tritium is also assumed to be unaffected by the treatment processes described.

{REDACTED FOR PUBLICATION Table 29.4-2. KNF Abatement Parameters (DFs)}

The above DFs are largely based on Reference [22]. These values will be updated with optimised, RR SMR-specific DFs and performance data obtained from treatment system manufacturers/vendors or commissioned studies.

{REDACTED FOR PUBLICATION}

Discharges to the environment

The RR SMR has been designed to minimise tritium production in the coolant. However, tritium still represents a limiting factor for recycling of treated aqueous effluent as there are currently no cost-effective techniques for removing tritium from large volumes of aqueous effluent. The concentration of tritium in the primary coolant is considered to reach steady state during the routine power operation. However, tritium may accumulate in the primary coolant due to releases from the fuel following in the event of fuel pin failures. Such accumulation may need to be managed through controlled reactor bleeds to KNF30 (following treatment in KNF20). Further, controlled bleed of the primary coolant to KNF30 may be required prior to RPV head lift during outages as an ALARP measure to minimise the exposure of outage workers to tritiated water vapours.

Treated effluents held in KNF30 tanks would normally be reused as demineralised water within the Reactor Island systems. On the occasions when treated effluent in KNF30 tanks cannot be reused or recycled within the Reactor Island and need to be discharged to the environment (e.g., for water balance purposes or in the event of fuel failure), the effluent will be released to the environment via a single point to the environment. Batches of treated effluent are stored in KNF30 tanks prior to release, thus discharges to the environment are not sensitive to plant operating modes. KNF30 tanks will be sampled prior to discharge and will only be discharged if the effluent meets permit conditions or other set limits. If the permit conditions and limits are not met, the tank content is returned to the KNF20 for further treatment.

The inventory and activity of radioactive effluent processed by KNF20 was determined from the output of a Mass Activity Balance (MAB) model.

Decay correction factors, and a correction factor to correct from per cycle to annual discharge activity were then applied to mass of liquid effluent (mass is determined from the effluent volumes reported in Table 29.4-1 by assuming a density of the liquid waste stream of **{REDACTED**

FOR PUBLICATION}) to determine the aqueous radioactive effluent discharged from KNF30 to the environment using the following relationship: **{REDACTED FOR PUBLICATION}**

Estimated activity of key radionuclides in the aqueous effluent discharge stream, derived using the above relationship are presented in Table 29.4-3. These take account of radioactive decay during effluent processing (around **{REDACTED FOR PUBLICATION}** hours) and assumes that the entire treated effluent inventory in one KNF30 tank **{REDACTED FOR PUBLICATION}**$m^3$is discharged. Radionuclides predicted to be discharged at activity levels below 100Bq are not included in the Table.

Table 29.4-3. Estimated Annual Discharge of Aqueous Radioactive Effluent from KNF30

Radionuclide	Annual discharge (Bq/y)	Radionuclide	Annual discharge (Bq/y)	Radionuclide	Annual discharge (Bq/y)
Ag-108m	3.04E+02	Cs-135	5.08E+00	Ni-63	1.62E+05
Ag-110m	2.89E+06	Cs-136	1.52E+08	Pr-143	1.62E+06
Ba-140	4.28E+07	Cs-137	1.12E+08	Ru-103	2.85E+07
C-14	5.07E+07	Fe-55	2.10E+07	Ru-106	4.97E+03
Ce-141	5.62E+05	Fe-59	8.25E+05	Sb-124	1.24E+06
Ce-144	1.58E+07	H-3	8.88E+10	Sb-125	1.06E+06
Cl-36	2.93E+00	I-131	4.88E+06	Sr-89	3.32E+06
Co-58	4.07E+06	Mn-54	2.36E+06	Sr-90	3.00E+05
Co-60	1.83E+06	Nb-94	5.24E+03	Y-91	2.02E+04
Cr-51	5.87E+06	Nb-95	1.13E+06	Zn-65	5.82E+06
Cs-134	2.39E+08	Ni-59	1.23E+03	Zr-95	1.33E+06

The above estimates are based on the interim PST derived as described in Section 29.3.3, along with information from the PCD aqueous waste management system design elements and/or conservative assumptions described in this Section. The data used to derive the Interim PST is obtained from nuclear power stations which have operated over different periods of time, and will therefore incorporate the range of radioactive discharges expected as a plant ages; including, for example, changes in pool chemistry and from expected events. The definitive PST will be refined to ensure lifetime operations are addressed in subsequent versions of this Chapter.

The processing and management of aqueous liquid effluent aims to reuse most treated effluent within the RR SMR, with the expectation that only small quantities of effluent unsuitable for reuse will be accumulated in KNF30 tanks for discharge to the environment. Discharges of aqueous liquid effluent are therefore expected only on a relatively infrequent basis. For the purposes of the preliminary assessment of potential radiological impacts via aqueous pathways, it is cautiously assumed that routine discharges of liquid radioactive effluent will be made. A detailed discharge strategy for RR SMR aqueous waste has not been developed at this stage, as it will necessarily need to consider plant operational requirements and site-specific environmental conditions unknown at this stage. However, as more fully discussed in Chapter 30, in a coastal setting such as that of the RR SMR Generic Site the total dose received will not differ significantly regardless of whether annual discharges are made as continuous discharge or as short-term releases.

29.4.3 Sources of radioactive gaseous effluent discharge

This section provides estimates of the gaseous radioactive effluent discharged from the RR SMR. This gaseous radioactive effluent will originate from one of the following sources:

The Processing and Treatment System for Gaseous Radioactive Effluent (GWTS) [KPL]

The KPL system uses nitrogen cover gas to purge interfacing systems handling reactor coolant, primary circuit effluent or make-up water. Hydrogen and volatile fission products (xenon and krypton) in vessel/tank headers are purged by the cover gas and collected as gaseous effluent. The key sources of these gases during power operation are the vacuum degasser in the processing & treatment system for liquid radioactive effluent (KNF) and Reactor Coolant Drains Tank (RCDT) in the CDS [KTA] [24].

Most of the nitrogen cover gas is recycled in a semi-closed loop. Excess gas during volume surges such as tank filling operations is directed to the delay beds where the fission product gases are abated through hold-up and decay. Thus, only small quantities of noble gases are expected to be released during the power operation mode.

The primary effluent storage tanks in KNF system are identified as the largest volume of gas (> **{REDACTED FOR PUBLICATION}** m^3 of free volume) and an increase in primary liquid effluent letdown (e.g. during reactor start-up transient) to KNF10 tanks will require some gaseous effluent volume to be discharged via the delay beds.

KNF degassing operations (prior to the reactor shutdown transient or following design basis fuel failure) are expected to generate significant quantities of hydrogen and fission product effluent. Similarly, the RCDT in CDS [KTA] is a significant source of gaseous effluent as it receives reactor coolant during operation, including the pressuriser steam bleeds.

Nuclear Heating, Ventilation and Air Conditioning (HVAC) system

The function of the nuclear HVAC [KL-] system is to extract and remove air contaminated with radioactivity from the atmospheres of controlled areas[6] and auxiliary areas of the Reactor Island. This covers the Containment Building, Auxiliary Building and the area around the Spent Fuel Pool (SFP, [FAB10]) and Refuelling Pool [FAF] areas. The radioactivity in the atmospheres of the Auxiliary and Containment Buildings originates from leakages of radioactive effluent from the primary circuit and – in the case of argon-41 from the Containment Building – neutron activation of the air around the RPV.

Extracted air is treated using High Efficiency Particulate Air (HEPA) filters to remove particulate radionuclides before discharge to the environment (together with radioactive gas stream from the GWTS [KPL]) via a gaseous emission stack.

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[6] Controlled areas, under Regulation 17(1) of the Ionising Radiation Regulations 2017 [57], are designated as such by the employer due to the need for special procedures to restrict exposure to ionising radiation, or where limits of 6 mSv/y effective dose, 15 mSv/y to the lens of the eye, or 150 mSv/y to skin are likely to be exceeded

Turbine Condenser Air Removal System (CARS)

Radioactive contamination of the secondary circuit arises through transfer of radioactivity from the primary circuit via small primary-to-secondary leaks within SG tubes. The leaked effluent is transported across the secondary circuit and non-condensable gases are eventually stripped by the (CARS)[MAJ] and discharged to the environment.

29.3.4 Estimates of gaseous radioactive effluent discharges from the GWTS [KPL]

Radioactive gases generated in the manner described are collected, cooled and dried to reduce the moisture content and improve the performance of the downstream treatment processes. The dried gases are then channelled to a catalytic recombiner for hydrogen and oxygen abatement, and then subjected to compression, further moisture removal and drying. A block flow diagram highlighting the key processes in managing gaseous radioactive effluent discharges is shown in Figure 29.4-3.

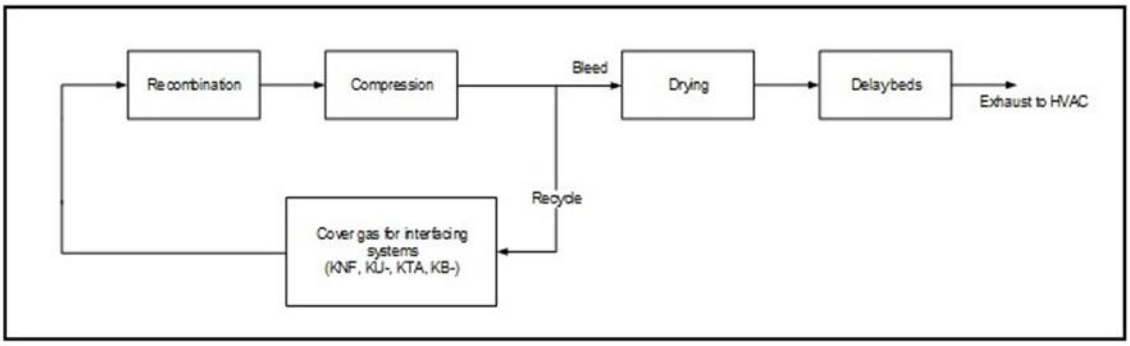

Figure 29.4-3 Block flow diagram for gaseous waste treatment system [KPL]

Following the above pre-treatment, the volatile fission products (primarily radioisotopes of krypton and xenon) are channelled through a series of activated charcoal delay beds which holds-up the gases to allow radioactive decay, before discharge to air via the gaseous emission stack [17]. The RR SMR GWTS will be designed to achieve hold-up times and decontamination factors aligned with Nuclear Safety Standards [25], as presented in Table 29.4-4. Full details of the GWTS [KPL] are provided in Reference [24].

{REDACTED FOR PUBLICATION Table 29.4-4. GWTS Abatement Parameters}

Tritium, radioiodine and aerosols in the gaseous effluent stream are assumed to be removed by the moisture removal steps and the guard bed (a bed of activated charcoal that acts as a sacrificial bed to protect the delay beds from contaminants or moisture carryover); and the quantities of these substances that reach the delay beds are therefore assumed to be negligible. In contrast, non-condensable activation products (particularly argon-41 and gas-phase carbon-14) present in the gaseous effluent stream are assumed to pass through the GWTS [KPL] relatively unaffected by any of the processing and treatment steps described.

There is evidence from an existing PWR (Sizewell B) that a fraction of carbon-14 produced over a fuel cycle may be accumulated in RCS delay beds via exchange with carbon-12 in the charcoal within the delay beds. This adsorbed carbon-14 is then available for release at a later date, either as the system is flushed following maintenance, or as a result of temperature/flow fluctuations during operations [23] [26] , such that gaseous carbon-14 discharge cannot be assumed to be continuous.

The RR SMR GWTS (KPL) is still in development; however, the intention is that the design and operation of the RR SMR should not result in a similar pattern of carbon-14 accumulation and delayed discharge observed at Sizewell B. Thus, for the current iteration of the quantification of radioactive discharges, it is assumed carbon-14 is not accumulated in the GWTS delay beds. The findings of the Sizewell B OPEX will be kept under review as the design develops.

{REDACTED FOR PUBLICATION}

Annual discharges of gaseous radioactive effluent from GWTS [KPL] to the environment during power operation and shutdown modes have been estimated using the following relationship:

{REDACTED FOR PUBLICATION}

Estimated discharges derived using the above relationship are presented in Table 29.4-5.

Table 29.4-5. Annual Discharge of Gaseous Radioactive Effluent from the GWTS [KPL]

Radionuclide	Power operation discharges (Bq/y)	Shutdown discharges (Bq/y)	Total discharges (Bq/y)
H-3	0.00E+00	0.00E+00	0.00E+00
C-14	0.00E+00	1.79E+11	1.79E+11
Ar-41	0.00E+00	2.25E+11	2.25E+11
Kr-85	9.81E+08	4.17E+10	4.27E+10

Radionuclide	Power operation discharges (Bq/y)	Shutdown discharges (Bq/y)	Total discharges (Bq/y)
Kr-85m	0.00E+00	1.52E+09	1.52E+09
Kr-87	0.00E+00	3.56E+02	3.56E+02
Kr-88	0.00E+00	7.51E+07	7.51E+07
Xe-131m	1.03E+08	4.08E+09	4.18E+09
Xe-133	0.00E+00	6.20E+10	6.20E+10
Xe-133m	0.00E+00	2.85E+06	2.85E+06

The noble gases with half-lives < 1 hour (krypton-89m, xenon-135, xenon-135m, xenon-137 and xenon-138) identified in the Interim PST (Table 29.3-1) will decay to insignificant levels in the delay beds due to considerably long hold-up times in the delay beds of 40hrs for krypton and 960hrs for xenon, and are thus excluded from Table 29.4-5. The delay beds are assumed to have no effect on carbon-14 and argon-41 discharges.

29.4.5 Estimates of gaseous radioactive effluent discharges from the nuclear HVAC [KL-] system

Design of the RR SMR nuclear HVAC [KL-] system is at an early stage, but some general design principles that will underpin the HVAC [KL-] design are presented.

The nuclear HVAC [KL-] will be designed to maintain ambient atmospheric conditions and a negative pressure across the areas it serves (within the Reactor Island) to prevent leakages and uncontrolled releases of gaseous radioactivity to the environment. The HVAC [KL-] will be configured so that exhausted air flows from areas with lowest potential for contamination to areas with higher potential for contamination in accordance with RGP [27]. The basic configuration of the system will likely be designed to provide continuous evacuation of air from buildings (e.g., containment building) at a nominal rate of 1 to 4 air changes per hour.

The key areas that are served by the nuclear HVAC [KL-], which have the potential to be significant sources of gaseous radioactive discharges to the atmosphere include:

1. The area around the Spent Fuel Pool [FAB10], Refuelling Pool [FAF] and other outage pools (comprising the Upender Pit [FAB40] and Refuelling Cavity [FAE], which account for the bulk of tritium discharges via the nuclear HVAC [KL-].

2. Other RCAs including the Containment Building (the main source of argon-41 discharges) and the Auxiliary Building (which contains various radioactive waste treatment systems and storage tanks), which also make important contribution to the HVAC [KL-] discharges. It is noted that a fraction of the HVAC discharge reported here from the Reactor Island HVAC [KL-] system may be discharged through an independent Turbine Building ventilation system in the final design; however, the total radioactive gaseous effluent discharged will not change.

Consistent with RGP for gaseous discharges from nuclear power plants [27] the HVAC [KL-] system will be fitted with HEPA filters upstream of the gaseous emission stack. The function of this filter is to remove particulates entrained in the air exhausted from different buildings served by the HVAC before discharge to the atmosphere via the emission stack. Typical removal efficiency and decontamination factor achievable for the HEPA filters is presented in Table 29.4-6 [28].

Table 29.4-6. HVAC Decontamination Factors

Filter type	Removal Efficiency	Decontamination Factors (DF)
HEPA filter	99.9%	1000

It is noted that HEPA filters can be designed or configured to achieve DFs greater than the values presented in Table 29.4-6 e.g., Reference [29]. The current design basis for the HVAC [KL-] incorporates activated charcoal filters that can be deployed to remove radioiodine from the HVAC discharge stream when required, for example under postulated fault conditions to minimise the release of radioiodine and the associated radiological consequences. The design requirements and specification of HEPA filters for the RR SMR will be established as the HVAC [KL-] system design progresses.

Contribution from the SFP and Refuelling Pool area HVAC Sub-system

The SFP [FAB10] provides storage for new, partially spent, fully spent and damaged fuel. Spent fuel is expected to be kept in fuel storage racks in the SFP [FAB10] for a period of 6-10 years, to allow sufficient cooling of the fuel before it is transferred to dry casks for long term storage. The current design capacity allows for up to 10 years of cooling following removal of fuel from the reactor core. The pool structure is made from stainless steel concrete composite (the same composition is used for all pools in the refuelling route) to provide adequate containment of water and shielding of the fuel. Reactivity control is achieved through geometrical spacing and the use of fixed neutron absorbers in the storage rack, with control rods providing additional control (not required for sub criticality) [17].

The SFP [FAB10] is connected to the Refuelling Pool [FAF] via the Fuel Transfer Channel [FCK] and Upender Pit [FAB40] – through which new, partially spent fuel and spent fuel is transferred between the two pools. The Refuelling Pool [FAF] is located inside containment next to the Refuelling Cavity [FAE] and contains racks for the storage of partially spent fuel assemblies fuel, as well as provision for the storage of RPV upper and lower internals, and RPV in-core instrumentation during refuelling. The current design is fitted with an ultrasonic cleaning station [FBC] for cleaning partially spent fuel [17]. Details of the current design configuration of the SFP and associated pools are presented in Reference [30]

Water quality in the SFP [FAB10], Upender Pit [FAB40] and Refuelling Pool [FAF] is maintained to the same standard as the primary coolant. This function is delivered by the Spent Fuel Pool Clean-up System (SFPCS) [FAK], which maintains the limits of chloride, fluoride and sulphate, and removes any FPs (mostly from leaking fuel) and dissolved crud. Also, the SFPCS [FAK] provides filters to cope with spalled particulate CPs and other debris [17]. Considerable, progressive reduction in the concentrations of these radionuclides in the SFP [FAB10] and associated pools (relative to the concentrations in the primary coolant at steady state power operation), is assumed due to the combined effects of losses via the SFPCS [FAK], radioactive decay, evaporative losses and dilution by demineralised make-up water. Tritium present in the water of the SFP [FAB10], Upender Pit [FAB40] and Refuelling Pool [FAF] is not removed by the SFPCS [FAK] processes.

The SFP [FAB10] and Refuelling Pool [FAF] are exposed to atmosphere for the during all operating modes and a small fraction of the water (and any entrained activity) in these pools evaporates continually. Thus, these pools constitute the primary source of discharges to the HVAC subsystem serving these pools during power operating mode.

During shutdown for refuelling, water in the SFP [FAB10], Upender Pit [FAB40], Refuelling Pool [FAF], Refuelling Cavity [FAE] and the RPV attain equilibrium due to mixing via the Fuel Transfer Channel [FCK] and the gate between the Refuelling Pool [FAF] and the flooded Refuelling Cavity [FAE] which is opened during refuelling. It is therefore assumed that the radionuclide composition and activity concentration across the SFP [FAB10] and the connected pools are the same during shutdown operation, and that the mixing of these pools of water result in the dilution of in the primary coolant activity. As the design progresses, these assumptions will be reviewed against the RR SMR design definition. The volumes of the Spent Fuel Pool and adjoining pools are given in Table 29.4-7.

{REDACTED FOR PUBLICATION}

{REDACTED FOR PUBLICATION Table 29.4-7. Pool Volumes}

{REDACTED FOR PUBLICATION}

Evaporation from SFP [FAB10] and connected pools and the partitioning/carryover of radionuclides in the water vapour makes a significant contribution to discharges of gaseous radionuclides to environment via the HVAC [KL-]. An initial analysis of pool evaporation has been performed to support the design of the Fuel Pool Supply System (FPSS) [FAT] [31], the results of which are summarised in Table 29.4-8

{REDACTED FOR PUBLICATION Table 29.4-8. Pool Evaporation Losses}

{REDACTED FOR PUBLICATION}

To estimate the activity concentration of radionuclides entrained in water vapour from the SFP [FAB10] and the associated pools, the following Partitioning Coefficients (PC) (i.e., ratio of the concentration of a radionuclide in the gas phase to its concentration in the liquid phase at equilibrium) are applied to the derived pools' source terms:

Table 29.4-9. Radionuclide Partition Coefficients (PC)

Radionuclide group	PC
Halogens	0.01
Other radionuclides	0.005

The PCs for halogens and other radionuclides were recommended for estimating the partitioning of radionuclides from the liquid to the steam phase in the SG tube [22], which operates at higher temperature and pressure conditions, and are therefore likely to be conservative for the SFP [FAB10] and associated pools. These assumptions will be reviewed as the design of the RR SMR SFP [FAB10] and HVAC [KL-] systems mature.

It is assumed that 90% of carbon-14 inventory in pools migrates to the gas phase based on analysis presented in Reference [23], which indicated that up to 10% of carbon-14 available for release is discharged in liquid effluents. Similarly, it is assumed that 10% of tritium migrates to the gas phases, based on observations made in Reference [22], which suggested that up to 90% of the total quantity of tritium in the primary coolant that is available for release is discharged via liquid pathways. These migration factors will be revisited during future iterations of the analysis.

Annual discharges to the environment of gaseous radioactive effluent entrained in vapours arising from the SFP [FAB10] and associated pools are estimated using the following relationship: **{REDACTED FOR PUBLICATION}**

As is evident from the above equations, the SFP [FAB10] and Refuelling Pool [FAF] are considered to contribute to HVAC [KL-] discharges during both power operations and shutdown operations, whereas the other outage pools (Upender Pit [FAB40] and Refuelling Cavity [FAE]) are only flooded during refuelling operations and contribute to HVAC [KL-] discharges only during shutdown/refuelling outages.

{REDACTED FOR PUBLICATION}

Predicted discharges of gaseous radioactive effluent from the SFP [FAB10], Refuelling Pool [FAF] and other outage pools [FAB40 and FAE] via the HVAC [KL-] system are presented are presented in Table 29.4-10.

{REDACTED FOR PUBLICATION Table 29.4-10. Annual Discharge of Gaseous Effluent from SFP and Associated Pools}

The estimated contribution of the SFP and associated pools to annual discharge of gaseous effluent incorporate the decontamination factors to reflect the downstream abatement provided by HEPA filters for particulate radionuclides (see Table 29.4-6).

Contribution from the Auxiliary Building, Containment Building and Turbine Building HVAC Sub-system

The contributions from the Auxiliary Building, Containment Building and the Turbine Building (collectively referred to as 'buildings of interest' elsewhere) to annual discharges of gaseous radioactive effluent via the nuclear HVAC have been estimated by applying leakage rates (representing various leakage mechanisms that lead to the transfer of radioactivity from the primary coolant to surrounding buildings and structures) to the RR SMR interim PST. These leakage rates have been taken from published data based on measurements from standard, older generation PWRs and, in a limited number of cases, engineering judgement [17]. This approach is considered to represent a suitably conservative case for the RR SMR design.

It is noted that the Turbine Building will have its own dedicated HVAC system; however, the information regarding emissions from this building is included here alongside the equivalent information for the Reactor Island buildings.

Annual discharge of gaseous radioactive effluent to the environment from HVAC systems serving buildings of interest have been estimated using the following relationships: **{REDACTED FOR PUBLICATION}**

As is evident from the above equations, the calculation approach considers both the primary coolant leakage rate into the buildings of interest and the effect of radionuclide partitioning between the liquid phase in the leaked volume and the vapor phase in the building atmosphere. The ensuing paragraphs provide an overview of key parameters in the above equations and parameter values adopted for estimating discharges originating from the identified buildings.

Coolant Leakage into Buildings

Radioactive gases present in the buildings of interest originate from leakages of coolant and steam from the primary circuit into the atmospheres of such buildings. A notable exception is argon-41 in the Containment Building, which is primarily generated by neutron activation of argon-40 present in the air around the RPV. The rates of coolant leakages from the primary circuit to the buildings of interest adopted for the purposes of this report are presented in Table 29.4-11

{REDACTED FOR PUBLICATION Table 29.4-11. Coolant Leakage Rates}

The above leakage rates were derived principally from observations and engineering judgement from standard PWRs [22] and have been used to estimate releases of radioactivity from the primary circuit to the identified buildings, applying appropriate adjustment factors (Table 29.4-12) to account for differences between RR SMR and the reference plant.

The fraction of noble gases released from the Containment Building per day in in Table 29.4-11 was derived using the following expression [22]:**{REDACTED FOR PUBLICATION}**

To account for differences between the reference plant (upon which the above leakage rates and other parameter values used in the calculations are based), the following adjustment factors for thermal power, coolant mass and containment volume were applied:

{REDACTED FOR PUBLICATION Table 29.4-12. Adjustment Factors}

The relevant design factors, leakage rate and radioactivity concentration estimates used to establish will be reviewed as the design progresses.

{REDACTED FOR PUBLICATION}

Argon-41 Release from Nuclear HVAC [KL-] Systems

Argon-41 discharged through the HVAC [KL-] system is formed by two mechanisms:

1. activation of stable argon-40 dissolved in the coolant. The inventory of argon-41 thus generated migrates to the buildings of interest through primary coolant leakages and is exhausted to the atmosphere (along with other radionuclides in the leaked effluent) by the HVAC [KL-] system.

2. neutron activation of stable argon-40 present in the air surrounding the RPV in the Containment Building. Argon-41 thus generated is extracted and released to the environment by the HVAC [KL-] system – either continuously or batchwise, when the containment is vented or purged, for example before maintenance outages.

Discharge of argon-41 generated via the first mechanism is estimated in the same manner as other radionuclides originating from the primary coolant leakages. A different approach is adopted for discharges of argon-41 from the containment air, based on the application of a scaling factor to the published argon-41 release rate presented in Table.29.4-13.

Table.29.4-13 Argon-41 Average Annual Release Rate

Parameter	Value
Argon-41 release rate (TBq/y)	1.26

The release rate in Table.29.4-13 is based on an average of reported releases of argon-41 from several PWR plants [22]. The predicted discharge of argon-41 from the RR SMR Containment Building by this mechanism has been calculated by applying the adjustment factor for containment volume presented in Table 29.4-12 to the argon-41 release rate. The total argon-41 annual discharge is determined by summing this value and the discharge of argon-41 calculated using Eq. 12.

Predicted discharge of gaseous radioactive effluent from the identified buildings via the HVAC [KL-] system is presented in Table 29.4-14.

{REDACTED FOR PUBLICATION Table 29.4-14. Annual Discharge of Gaseous Effluent from Buildings}

Aggregated discharges from nuclear HVAC sources

Annual discharges of gaseous radioactive effluents via the nuclear HVAC system from all the identified structures and buildings of interest (including SFP and associated pools), derived using the equations, parameter values and assumptions described in the preceding section, are presented in Table 29.4-15.

Table 29.4-15. Estimated Annual Discharges of Gaseous Effluents via the Nuclear HVAC System

Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)	Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)
Ag-110m	2.71E+03	9.36E+04	9.63E+04	Nb-95	5.89E+03	9.03E+01	5.98E+03
Ar-41	5.61E+11	1.13E+09	5.75E+11	Nb-97	5.73E+03	8.79E+01	5.82E+03
Ba-140	2.58E+05	3.96E+03	2.62E+05	Ni-59	9.60E+01	3.04E+02	4.00E+02
Br-84	1.18E+09	2.07E+08	1.39E+09	Ni-63	1.27E+04	4.02E+04	5.29E+04
C-14	9.14E+08	1.40E+07	9.28E+08	Pr-143	9.61E+03	1.47E+02	9.75E+03
Ce-141	2.95E+03	4.52E+01	2.99E+03	Rb-88	2.41E+04	7.34E+03	3.14E+04
Ce-143	6.20E+04	9.51E+02	6.29E+04	Rb-89	2.65E+04	8.07E+03	3.45E+04
Ce-144	7.61E+04	1.17E+03	7.73E+04	Ru-103	1.47E+05	2.25E+03	1.49E+05
Co-58	1.78E+04	2.08E+06	2.10E+06	Ru-105	5.30E+04	8.14E+02	5.38E+04
Co-60	1.95E+03	4.28E+04	4.48E+04	Sb-122	1.03E+03	9.37E+04	9.48E+04
Cr-51	2.38E+04	2.34E+05	2.58E+05	Sb-124	8.22E+02	3.90E+04	3.99E+04

Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)	Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)
Cs-134	1.18E+04	4.35E+03	1.61E+04	Sb-125	9.32E+03	6.62E+03	1.59E+04
Cs-136	1.85E+03	9.20E+02	2.77E+03	Sr-89	2.75E+03	4.21E+01	2.79E+03
Cs-137	7.51E+03	2.30E+03	9.82E+03	Sr-90	2.34E+02	3.59E+00	2.38E+02
Cs-138	5.18E+05	2.34E+04	5.41E+05	Sr-91	2.52E+04	3.87E+02	2.56E+04
Fe-55	2.65E+04	3.03E+06	3.05E+06	Sr-92	3.98E+05	6.11E+03	4.04E+05
Fe-59	1.10E+03	1.26E+05	1.27E+05	Tc-99m	4.84E+02	7.43E+00	4.92E+02
H-3	3.81E+10	5.85E+08	3.87E+10	Te-131	2.87E+05	4.40E+03	2.91E+05
I-131	4.80E+07	1.69E+07	6.49E+07	Te-132	3.52E+04	5.40E+02	3.57E+04
I-132	5.04E+08	8.98E+07	5.94E+08	Te-133m	3.44E+05	5.28E+03	3.49E+05
I-133	2.24E+08	2.66E+07	2.50E+08	Te-134	6.24E+05	9.58E+03	6.34E+05
I-134	8.63E+08	1.05E+08	9.68E+08	Xe-131m	1.79E+09	7.88E+07	1.87E+09
I-135	4.64E+08	4.99E+07	5.14E+08	Xe-133	5.19E+11	2.29E+10	5.42E+11
K-42	2.80E+07	4.30E+05	2.85E+07	Xe-133m	1.93E+10	1.06E+09	2.04E+10
Kr-85	1.68E+09	7.50E+07	1.75E+09	Xe-135	1.58E+11	6.34E+09	1.64E+11
Kr-85m	2.05E+10	1.10E+09	2.16E+10	Xe-135m	8.78E+10	4.42E+09	9.22E+10
Kr-87	3.48E+10	1.87E+09	3.67E+10	Xe-138	1.06E+11	7.23E+09	1.13E+11
Kr-88	4.11E+10	2.29E+09	4.34E+10	Y-91m	1.66E+04	2.55E+02	1.68E+04
La-140	5.43E+05	8.32E+03	5.51E+05	Y-92	3.64E+05	5.59E+03	3.70E+05
Mn-54	3.55E+03	2.60E+04	2.95E+04	Zn-65	6.22E+03	1.37E+05	1.43E+05
Mn-56	2.02E+04	1.47E+05	1.68E+05	Zr-95	6.66E+03	1.02E+02	6.76E+03
Mo-99	9.42E+02	1.44E+01	9.56E+02	Zr-97	1.21E+04	1.86E+02	1.23E+04
Na-24	1.05E+05	1.61E+03	1.06E+05				

The estimated annual releases in Table 29.4-15 incorporate the decontamination factor presented in Table 29.4-6. Reduction in the inventory of radionuclides through radioactive decay has not been considered – except for noble gases (other than argon-41) released from the containment atmosphere (as described in Eq. 12] and very short-lived radionuclides (< 5min) which will decay to low levels during transit through HVAC ducts (based on HVAC air removal rates of 1-4 building volumes per hour). Similarly, a cautious DF of 1000 was adopted for the HEPA filter in the HVAC for the purposes of estimating discharges. A cautious DF of 1,000 as recommended in IAEA TRS 325 [32] is applied as an interim measure, in recognition of the current stage of the design of the RR SMR. It is anticipated a DF in excess of 1,000 is likely to be achieved and industry standard HPEA filters, which are readily utilised throughout the nuclear industry will be used and designed to deliver an optimised performance. Radionuclides with very low annual discharged rates (<100Bq/y) have not been presented in Table 29.4-14.

As mentioned in previous sections, the RR SMR HVAC design and definitive PST are currently in development – more representative underpinning design information is expected as the design progresses. The current estimates will therefore be reviewed and updated as the design of the RR SMR progresses.

{REDACTED FOR PUBLICATION}

29.4.6 Estimates of gaseous radioactive effluent discharges from the Condenser Air Removal System (CARS) [MAJ]

An overview of the mechanism and leakage rates for primary-to-secondary circuit leakage are described in Section 29.4.5. In the aftermath of this event, leaked radionuclides are transported and distributed across the secondary circuit. Condensable radionuclides (e.g., tritium) and particulates are condensed and retained within the aqueous phase of the Condenser, whilst volatile radionuclides partition to the gas phase and are eventually stripped and ejected to the atmosphere by the CARS [MAJ].

Design of the RR SMR CARS [MAJ] is currently in progress and the following assumptions have been made to facilitate estimation of potential discharges from this system.

1. The entire inventory of noble gases (including argon-41) and gas-phase carbon-14 (i.e., 90% of the total carbon-14 inventory) in the leaked effluent are discharged to the environment by CARS [MAJ]

2. The inventory of particulate and solubilised radionuclides (including tritium) in the leaked effluent are condensed and retained in the Condenser aqueous phase

3. A fraction of the transferred radioiodine inventory partitions to the secondary circuit gas phase according to the partition coefficients listed in Table 29.4-9 .

4. CARS [MAJ] will operate when condenser is in operation and there may be situations where condenser operation is required but electrical power generation is not needed.

5. Discharges from the CARS [MAJ] are released to the environment without abatement.

Annual discharges of noble gases, carbon-14 and radioisotopes of iodine to the atmosphere from CARS [MAJ] have been calculated using the following relationship:

{REDACTED FOR PUBLICATION}

Estimated annual discharges derived using the above relationship are presented in Table 29.4-16.

Table.29.4-16. Annual Discharge of Gaseous Radionuclides via CARS

Radionuclide	Annual discharge (Bq/y)	Radionuclide	Annual discharge (Bq/y)
Ar-41	4.11E+09	Kr-85m	7.32E+08
C-14	5.06E+07	Kr-87	1.24E+09
I-129	2.80E-04	Kr-88	1.46E+09
I-131	2.66E+06	Xe-131m	6.38E+07
I-132	2.79E+07	Xe-133	1.85E+10
I-133	1.24E+07	Xe-133m	6.88E+08
I-134	4.78E+07	Xe-135	5.63E+09
I-135	2.57E+07	Xe-135m	3.13E+09
Kr-85	5.98E+07	Xe-138	3.77E+09

As mentioned in previous sections, a definitive PST is currently in development and the design of the RR SMR CARS [MAJ] is in progress. The discharge estimates presented in Table 29.4-16 will be reviewed and updated accordingly.

{REDACTED FOR PUBLICATION}

29.4.7 Summary of predicted annual discharge of aqueous and gaseous radioactive effluents

A summary of the predicted annual discharges for aqueous and gaseous discharges to the environment from the RR SMR is presented in Table 29.4-17 and Tale 29.4-18 below. Table 29.4-18 is an aggregation of the predicted discharges from the GWTS [KPL], HVAC [KL] and CARS [MAJ].

Table 29.4-17. Predicted Annual Discharge of Aqueous Radioactive Effluent from LMDS [KNF30]

Radionuclide	Annual discharge (Bq/y)	Radionuclide	Annual discharge (Bq/y)	Radionuclide	Annual discharge (Bq/y)
Ag-108m	3.04E+02	Cs-135	5.08E+00	Ni-63	1.62E+05
Ag-110m	2.89E+06	Cs-136	1.52E+08	Pr-143	1.62E+06
Ba-140	4.28E+07	Cs-137	1.12E+08	Ru-103	2.85E+07
C-14	5.07E+07	Fe-55	2.10E+07	Ru-106	4.97E+03
Ce-141	5.62E+05	Fe-59	8.25E+05	Sb-124	1.24E+06
Ce-144	1.58E+07	H-3	8.88E+10	Sb-125	1.06E+06
Cl-36	2.93E+00	I-131	4.88E+06	Sr-89	3.32E+06
Co-58	4.07E+06	Mn-54	2.36E+06	Sr-90	3.00E+05
Co-60	1.83E+06	Nb-94	5.24E+03	Y-91	2.02E+04
Cr-51	5.87E+06	Nb-95	1.13E+06	Zn-65	5.82E+06
Cs-134	2.39E+08	Ni-59	1.23E+03	Zr-95	1.33E+06

Table 29.4-18. Predicted Annual Discharge of Gaseous Radioactive Effluent from GWTS [KPL], HVAC [KL-] & CARS [MAJ]

Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)	Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)
Ag-110m	2.71E+03	9.36E+04	9.63E+04	Nb-95	5.89E+03	9.03E+01	5.98E+03
Ar-41	5.65E+11	2.26E+11	8.04E+11	Nb-97	5.73E+03	8.79E+01	5.82E+03
Ba-140	2.58E+05	3.96E+03	2.62E+05	Ni-59	9.60E+01	3.04E+02	4.00E+02
Br-84	1.25E+09	2.07E+08	1.45E+09	Ni-63	1.27E+04	4.02E+04	5.29E+04
C-14	9.65E+08	1.79E+11	1.80E+11	Pr-143	9.61E+03	1.47E+02	9.75E+03
Ce-143	6.20E+04	9.51E+02	6.29E+04	Rb-88	2.41E+04	7.34E+03	3.14E+04
Ce-144	7.61E+04	1.17E+03	7.73E+04	Rb-89	2.65E+04	8.07E+03	3.45E+04
Co-58	1.78E+04	2.08E+06	2.10E+06	Ru-103	1.47E+05	2.25E+03	1.49E+05

Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)	Radio-nuclide	Power Operation (Bq/y)	Shutdown (Bq/y)	Total (Bq/y)
Co-60	1.95E+03	4.28E+04	4.48E+04	Ru-105	5.30E+04	8.14E+02	5.38E+04
Cr-51	2.38E+04	2.34E+05	2.58E+05	Sb-122	1.03E+03	9.37E+04	9.48E+04
Cs-134	1.18E+04	4.35E+03	1.61E+04	Sb-124	8.22E+02	3.90E+04	3.99E+04
Cs-136	1.85E+03	9.20E+02	2.77E+03	Sb-125	9.32E+03	6.62E+03	1.59E+04
Cs-137	7.51E+03	2.30E+03	9.82E+03	Sr-89	2.75E+03	4.21E+01	2.79E+03
Cs-138	5.18E+05	2.34E+04	5.41E+05	Sr-90	2.34E+02	3.59E+00	2.38E+02
Fe-55	2.65E+04	3.03E+06	3.05E+06	Sr-91	2.52E+04	3.87E+02	2.56E+04
Fe-59	1.10E+03	1.26E+05	1.27E+05	Sr-92	3.98E+05	6.11E+03	4.04E+05
H-3	4.02E+10	5.85E+08	4.08E+10	Tc-99m	4.84E+02	7.43E+00	4.92E+02
I-131	5.06E+07	1.69E+07	6.76E+07	Te-131	2.87E+05	4.40E+03	2.91E+05
I-132	5.32E+08	8.98E+07	6.22E+08	Te-132	3.52E+04	5.40E+02	3.57E+04
I-133	2.36E+08	2.66E+07	2.63E+08	Te-133m	3.44E+05	5.28E+03	3.49E+05
I-134	9.11E+08	1.05E+08	1.02E+09	Te-134	6.24E+05	9.58E+03	6.34E+05
I-135	4.90E+08	4.99E+07	5.40E+08	Xe-131m	1.96E+09	4.16E+09	6.11E+09
K-42	2.80E+07	4.30E+05	2.85E+07	Xe-133	5.37E+11	8.49E+10	6.22E+11
Kr-85	2.72E+09	4.18E+10	4.45E+10	Xe-133m	2.00E+10	1.06E+09	2.10E+10
Kr-85m	2.13E+10	2.62E+09	2.39E+10	Xe-135	1.64E+11	6.34E+09	1.70E+11
Kr-87	3.61E+10	1.87E+09	3.79E+10	Xe-135m	9.09E+10	4.42E+09	9.53E+10
Kr-88	4.26E+10	2.36E+09	4.49E+10	Xe-138	1.10E+11	7.23E+09	1.17E+11
La-140	5.43E+05	8.32E+03	5.51E+05	Y-91m	1.66E+04	2.55E+02	1.68E+04
Mn-54	3.55E+03	2.60E+04	2.95E+04	Y-92	3.64E+05	5.59E+03	3.70E+05
Mn-56	2.02E+04	1.47E+05	1.68E+05	Zn-65	6.22E+03	1.37E+05	1.43E+05
Mo-99	9.42E+02	1.44E+01	9.56E+02	Zr-95	6.66E+03	1.02E+02	6.76E+03
Na-24	1.05E+05	1.61E+03	1.06E+05	Zr-97	1.21E+04	1.86E+02	1.23E+04

The totalled annual discharges data presented in Table 29.4-18 are based on a rolling 12-month average that encompasses the shutdown period and therefore represent the higher estimate of discharges expected to occur in any 12-month period. **{REDACTED FOR PUBLICATION}**

29.5 Estimation of Monthly Discharges

29.5.1 Determination of significant radionuclides

Monthly estimates of aqueous and gaseous radioactive discharges to the environment are provided to support future permitting activities for the RR SMR [1]:

1. *on an individual radionuclide basis for significant radionuclides*

2. *on a group basis (for example 'total alpha' or 'total beta') for other radionuclides*

3. *via each discharge point and discharge route*

Significant radionuclides are defined as those which:

1. *have a radiological impact on people or non-human species*

2. *discharge high quantities of radioactivity*

3. *have long half-lives, may persist or accumulate (or both) in the environment, and may contribute significantly to collective dose*

4. *are indicators of facility performance and process control*

The radionuclide selection should be consistent with Recommendation 2004/2/Euratom [2].

The Environment Agency has published a guidance document setting out their criteria for setting limits on the discharge of radioactive effluents from nuclear sites [33]. This guidance document establishes the basis for determination of significant radionuclides (i.e., radionuclides for which annual discharge limits will be set), which includes the following conditions:

a) *are significant in terms of radiological impact on people (that is, the dose to the most exposed group at the proposed limit exceeds 1μSv/y);*

b) *are significant in terms of radiological impact on non-human species (this only needs to be considered where the impact on reference organisms from the discharges of all radionuclides at the proposed limits exceeds 40μGy/h);*

c) *are significant in terms of the quantity of radioactivity discharged (that is, the discharge of a radionuclide exceeds 1TBq/y);*

d) *may contribute significantly to collective dose (this only needs to be considered where the collective dose truncated at 500 years from the discharges of all radionuclides at the proposed limits exceeds 1 man-sievert per year to any of the UK, European or World populations);*

e) *are constrained under national or international agreements or is of concern internationally;*

f) *are indicators of plant performance, if not otherwise limited on the above criteria; and*

g) *for the appropriate generic categories from the RSR Pollution Inventory (eg "alpha particulate" and "beta/gamma particulate" for discharges to air) to limit any radionuclides not otherwise covered by the limits set on the above criteria.*

A screening assessment of potential doses to members of the public and dose rates to non-human species was performed using the Environment Agency's Initial Radiological Assessment Tool (IRAT), based on the annual discharge data presented in Table 29.4-17 and Table 29.4-18. Default environmental parameters embedded in the IRAT tool were used for the Stage 1 assessment. The Stage 2 assessment used the coastal volumetric exchange rate of 100 m^3/s selected for the GSD, and the effective release height for gaseous effluent of **{REDACTED FOR PUBLICATION}**, **{REDACTED FOR PUBLICATION}**. The full details and results of these assessments and the assumptions used in modelling radiological impact are provided in Chapter 30 of the E3S Case.

The Stage 2 assessment established that:

1. A total dose of around 0.3μSv/y to the worst affected member of a fishing family and 7μSv/y to the worst affected local inhabitant were calculated for aqueous discharges to sea and gaseous discharges to atmosphere, respectively. The radioisotopes of caesium (53%), zinc-65 (13%) and carbon-14 (18%) accounted for over 80% of the dose from aqueous discharges, whilst the dose arising from gaseous discharges is largely attributable to the intake of carbon-14 (>90%)

2. The radioactive effluent discharges RR SMR discharges result in doses which are below the threshold of significance for conditions a), b), c), or e). Condition d) (collective dose) has not been formally assessed, but is not anticipated to be of concern, given the relatively low doses calculated for a) and b) using the conservative IRAT model. Collective dose for aqueous discharges will be assessed once sufficient information on the PST and RR SMR design is available.

3. None of the radionuclides predicted to be present in gaseous effluent discharges were found to satisfy conditions b), c) or e). Carbon-14 and argon-41 gave rise to a dose of around 13μSv/y (to an infant) and 2.6μSv/y (to an adult), respectively, using default IRAT model parameters. These doses decreased to 3.2μSv/y for carbon-14 and 0.1μSv/y for argon-41 when an effective release height of 15m is applied. Carbon-14 therefore satisfies condition a). As for aqueous effluent discharges, collective dose has not been formally assessed at this stage of the process but will be evaluated as the design matures.

{REDACTED FOR PUBLICATION}

A review of the annual discharge limits for Sizewell B (an operating PWR power station) and Hinkley Point C (a proposed PWR power station) was also carried out:

1. The 2020 Radioactivity in Food and the Environment (RIFE) report [36] indicates that annual limits are set for noble gases, particulate beta, tritium, carbon-14 and iodine-131 (in gaseous effluent) and tritium, caesium-137 and 'other radionuclides' (in aqueous effluent) discharged from Sizewell B.

2. In its decision document for the Hinkley Point C nuclear power station radioactive substances activity permit [36], the Environment Agency set annual limits on the discharge of tritium, carbon-14, noble gases, iodine-131 and other fission & activation products in

gaseous discharges; and of tritium, carbon-14, cobalt-60, caesium-137 and other fission & activation products in aqueous discharges.

In addition, the GDA submission for UK HPR1000 – the most recent nuclear power plant design to successfully complete the GDA process – identified tritium, carbon-14, xenon-133, xenon-135 and other gamma / beta emitters (in gaseous discharges) and tritium, carbon-14 and other gamma emitters (in aqueous discharges) as significant radionuclides from that power plant [37].

Table 29.5-1 presents a summary of radionuclides identified for consideration as significant radionuclides.

Table 29.5-1. Significant Radionuclides

Criterion	Aqueous	Gaseous
Public dose at proposed limit > 1 μSv/y	--	Carbon-14
Dose rate to non-human species at proposed limits > 40μGy/h	--	--
Discharge exceeds 1TBq/y	--	--
Collective dose at proposed limits > 1 man-Sv /y to any population	Not assessed	Not assessed
Of international concern	Not known	Not known
Indicator of plant performance	Tritium, caesium-137	Tritium, xenon-133
Established/ proposed limits for PWRs in the UK	Tritium, carbon-14, cobalt-60, caesium-137, other radionuclides	Tritium, carbon-14, noble gases, iodine-131, xenon-133, xenon-135 and other fission & activation products

Taking account of the predicted discharges from the RR SMR, the evidence presented in the preceding paragraphs and the key radionuclides listed in Annex 1 of 2004/2/Euratom [2], the following are considered to be the significant radionuclides predicted to be discharged from the RR SMR:

1. Aqueous discharges: tritium and caesium-137

2. Gaseous discharges: tritium, carbon-14, iodine-131 and xenon-133.

It is noted that the definitive source term for the RR SMR is currently in production, and that the various aqueous and gaseous effluent abatement systems are in development. The identified significant radionuclides are therefore subject to review once the definitive PST and further design information are available.

{REDACTED FOR PUBLICATION}

29.5.2 Estimation of monthly discharge of aqueous and gaseous radioactive effluents to the environment

The data underpinning the predicted annual discharges from the RR SMR presented in the previous Section are not sufficiently detailed to allow the derivation of monthly discharges for the RR SMR. Further, the design of relevant systems that have a bearing on the generation and discharge of radioactive effluent, and frequency of discharges to the environment (including related plant operating philosophies) are currently in development. It is therefore challenging to provide precise estimates of monthly discharges from the RR SMR.

Nevertheless, considering the design objective that the RR SMR operates with an overall capacity factor of > 0.90 [17], it can be assumed that the plant can be operated with minimal variation in routine discharges to the environment during power operations on the basis that:

1. Aqueous effluents are stored in KNF30 tanks for reuse as demineralised water; the discharge of this effluent stream to the environment is expected to be infrequent and a reasonable bounding case is for the total inventory of radionuclides accumulated in a tank to be released over a few hours.

2. For gaseous effluent discharges, it is considered that discharges over the course of the **{REDACTED FOR PUBLICATION}** power operation mode will be relatively uniform, with a spike in the discharge of some radionuclides during the **{REDACTED FOR PUBLICATION}** shutdown period in the last (i.e., 18th) month of the operating cycle. The bounding case for gaseous discharges is therefore expected to occur in the 18th month of the operating cycle **{REDACTED FOR PUBLICATION}**.

The maximal monthly release of significant radionuclides in aqueous and gaseous discharges from the RR SMR is estimated using the following relationships:

{REDACTED FOR PUBLICATION}

The estimated monthly discharge of significant gaseous radionuclides derived using the above relationship are presented in Table 29.5-2 and Table 29.5-3

Table 29.5-2 Maximal Monthly Discharge of Significant Aqueous Radionuclides #

Radionuclide	Annual discharge (Bq/y)	Monthly discharge (Bq/m)#
H-3	8.88E+10	8.88E+10
Cs-137	1.12E+08	1.12E+08

#Based on the assumption that annual discharge is released in one month

Table 29.5-3 Indicative Monthly Discharge of Significant Gaseous Radionuclides

Month	Tritium discharge (Bq/m)	Carbon-14 discharge (Bq/m)	Iodine-131 discharge (Bq/m)	Xenon-133 discharge (Bq/m)
1	3.24E+09	7.77E+07	4.08E+06	4.33E+10
2	3.24E+09	7.77E+07	4.08E+06	4.33E+10
3	3.24E+09	7.77E+07	4.08E+06	4.33E+10
4	3.24E+09	7.77E+07	4.08E+06	4.33E+10
5	3.24E+09	7.77E+07	4.08E+06	4.33E+10

Month	Tritium discharge (Bq/m)	Carbon-14 discharge (Bq/m)	Iodine-131 discharge (Bq/m)	Xenon-133 discharge (Bq/m)
6	3.24E+09	7.77E+07	4.08E+06	4.33E+10
7	3.24E+09	7.77E+07	4.08E+06	4.33E+10
8	3.24E+09	7.77E+07	4.08E+06	4.33E+10
9	3.24E+09	7.77E+07	4.08E+06	4.33E+10
10	3.24E+09	7.77E+07	4.08E+06	4.33E+10
11	3.24E+09	7.77E+07	4.08E+06	4.33E+10
12	3.24E+09	7.77E+07	4.08E+06	4.33E+10
13	3.24E+09	7.77E+07	4.08E+06	4.33E+10
14	3.24E+09	7.77E+07	4.08E+06	4.33E+10
15	3.24E+09	7.77E+07	4.08E+06	4.33E+10
16	3.24E+09	7.77E+07	4.08E+06	4.33E+10
17	3.24E+09	7.77E+07	4.08E+06	4.33E+10
18	1.94E+09	1.79E+11	1.86E+07	1.03E+11
Total	5.70E+10	1.80E+11	8.79E+07	8.39E+11

The predicted monthly discharges will be re-estimated on completion of the review of predicted discharges and review the list of significant radionuclides, once a definitive RR SMR PST is available.

29.6 Proposed Discharge Limits

29.6.1 Introduction

Prospective operators of new nuclear power station operators applying for an environmental permit under the RSR regime are required to provide proposals for annual site limits (on a rolling 12-month basis) for gaseous and aqueous discharges, supported by a description of the justification of the requested limits [38]. Annual limits are set to ensure that BAT is applied to minimise discharges to the environment and that the dose to members of the public is below statutory limits and constraints, and to ensure adequate protection of the environment. An estimate of proposed annual site limits, based on the quantified discharges at PCD is given in this section.

Annual limits are normally established for selected radionuclides determined using set criteria published in Reference [33]. These criteria have been applied to predicted aqueous and gaseous discharges from the RR SMR plant and the outcome supplemented with analyses of discharge limits that were proposed in previous GDA submissions and those that have been imposed on permitted or operating nuclear power stations. Details of these analyses and the resultant significant radionuclides identified for the RR SMR are presented in Section 29.5. The following Sections propose limits on discharges of the RR SMR significant radionuclides including an account of how the limits have been determined.

29.6.2 Approach for evaluating proposed limits

In proposing appropriate limits on discharges of radioactive effluents to the environment, account is taken of both the predicted discharges under normal operating conditions (including variability associated with different operating modes including outages) and the contributions from infrequent, random events expected to occur over the lifetime of the facility [33].

The annual limits on discharges of the significant radionuclides identified in Table 29.5-1 to the environment from the RR SMR have been determined using the following relationship:

{REDACTED FOR PUBLICATION}

As described in Section 29.3, the interim PST has been derived from OPEX data from operational PWRs, with adjustments for RR SMR-specific design factors. The OPEX data are considered to include the contributions from expected events typical of PWRs. The predicted RR SMR annual discharges presented in Table 29.4-17 and Table 29.4-18 are therefore taken to incorporate both routine discharges and the contributions from expected events.

Determination of headroom factor

In setting annual limits, headroom factors are applied to predicted discharges of radioactive effluent to ensure that operators of permitted facilities can comply with the proposed limits without unduly affecting their ability to operate. **{REDACTED FOR PUBLICATION}**

The most recent GDA submissions for new nuclear power stations adopted a statistical approach for determining appropriate headroom factors [37], [39]. This approach is based on the application of a one-sided normal distribution at 99.9th percentile confidence interval to the operational data upon which the discharges were estimated. It is noted that HPR1000 and UK-ABWR had access to OPEX from sister sites of the same or similar design.

The RR SMR interim PST uses OPEX from a range of PWRs which has been scaled to thermal output and adjusted for the KOH chemistry regime, using a range of factors and modelling [9].

The OPEX data used to determine the RR SMR interim PST, and thus the interim predicted discharges is not sufficiently detailed to allow adoption of the statistical approach described above. Thus, an alternative approach based on recommendation in the Environment Agency's limit setting report [40] is adopted. The report recommends the use of "worst case plant discharge" (WCPD) approach which takes account of factors that may lead to variability in predicted discharges including expected events, power output and plant ageing, among other things. The report recommends that the WCPD for new plants should be a factor of 2 times the best estimate of discharges of radioactive waste.

Annual limits for radioactive effluent discharges determined from the RR SMR interim PST have therefore been determined by determining the WCPD using Eq. 17 above, with a value of 2 ascribed to the headroom factor, HF. The proposed annual limits are presented in Section 29.6.3.

29.6.3 Proposed Limits

Table 29.6-1 and Table 29.6-2 present estimates of discharges of radionuclides in aqueous and gaseous effluent at annual limits, predicted to arise from the RR SMR plant.

Table 29.6-1. Predicted Discharges of Aqueous Radionuclides at Annual Limits

Radionuclide	Predicted annual discharges (Bq/y)	Discharges at annual limits (Bq/y)	Radionuclide	Predicted annual discharges (Bq/y)	Discharges at annual limits (Bq/y)
Ag-108m	3.04E+02	6.08E+02	I-131	4.88E+06	9.76E+06
Ag-110m	2.89E+06	5.78E+06	Mn-54	2.36E+06	4.72E+06
Ba-140	4.28E+07	8.56E+07	Nb-94	5.24E+03	1.05E+04
C-14	5.07E+07	1.01E+08	Nb-95	1.13E+06	2.26E+06
Ce-141	5.62E+05	1.12E+06	Ni-59	1.23E+03	2.46E+03
Ce-144	1.58E+07	3.16E+07	Ni-63	1.62E+05	3.24E+05
Cl-36	2.93E+00	5.86E+00	Pr-143	1.62E+06	3.24E+06
Co-58	4.07E+06	8.14E+06	Ru-103	2.85E+07	5.70E+07
Co-60	1.83E+06	3.66E+06	Ru-106	4.97E+03	9.94E+03
Cr-51	5.87E+06	1.17E+07	Sb-124	1.24E+06	2.48E+06
Cs-134	2.39E+08	4.78E+08	Sb-125	1.06E+06	2.12E+06
Cs-136	1.52E+08	3.04E+08	Sr-89	3.32E+06	6.64E+06
Cs-137	1.12E+08	2.24E+08	Sr-90	3.00E+05	6.00E+05
Fe-55	2.10E+07	4.20E+07	Y-91	2.02E+04	4.04E+04
Fe-59	8.25E+05	1.65E+06	Zn-65	5.82E+06	1.16E+07
H-3	8.88E+10	1.78E+11	Zr-95	1.33E+06	2.66E+06

Table 29.6-2. Predicted Discharges of Gaseous Radionuclides at Annual Limits

Radionuclide	Predicted annual discharges (Bq/y)	Discharges at annual limits (Bq/y)	Radionuclide	Predicted annual discharges (Bq/y)	Discharges at annual limits (Bq/y)
Ag-110m	9.63E+04	1.93E+05	Nb-95	5.98E+03	1.20E+04
Ar-41	8.04E+11	1.61E+12	Nb-97	5.82E+03	1.16E+04
Ba-140	2.62E+05	5.24E+05	Ni-59	4.00E+02	8.01E+02
Br-84	1.45E+09	2.91E+09	Ni-63	5.29E+04	1.06E+05
C-14	1.80E+11	3.59E+11	Pr-143	9.75E+03	1.95E+04
Ce-143	6.29E+04	1.26E+05	Rb-88	3.14E+04	6.29E+04
Ce-144	7.73E+04	1.55E+05	Rb-89	3.45E+04	6.90E+04
Co-58	2.10E+06	4.20E+06	Ru-103	1.49E+05	2.98E+05
Co-60	4.48E+04	8.96E+04	Ru-105	5.38E+04	1.08E+05
Cr-51	2.58E+05	5.15E+05	Sb-122	9.48E+04	1.90E+05
Cs-134	1.61E+04	3.22E+04	Sb-124	3.99E+04	7.97E+04
Cs-136	2.77E+03	5.54E+03	Sb-125	1.59E+04	3.19E+04
Cs-137	9.82E+03	1.96E+04	Sr-89	2.79E+03	5.58E+03
Cs-138	5.41E+05	1.08E+06	Sr-90	2.38E+02	4.76E+02
Fe-55	3.05E+06	6.11E+06	Sr-91	2.56E+04	5.12E+04
Fe-59	1.27E+05	2.55E+05	Sr-92	4.04E+05	8.08E+05
H-3	4.08E+10	8.16E+10	Tc-99m	4.92E+02	9.83E+02
I-131	6.76E+07	1.35E+08	Te-131	2.91E+05	5.83E+05
I-132	6.22E+08	1.24E+09	Te-132	3.57E+04	7.15E+04
I-133	2.63E+08	5.25E+08	Te-133m	3.49E+05	6.99E+05
I-134	1.02E+09	2.03E+09	Te-134	6.34E+05	1.27E+06
I-135	5.40E+08	1.08E+09	Xe-131m	6.11E+09	1.22E+10
K-42	2.85E+07	5.69E+07	Xe-133	6.22E+11	1.24E+12
Kr-85	4.45E+10	8.90E+10	Xe-133m	2.10E+10	4.21E+10
Kr-85m	2.39E+10	4.78E+10	Xe-135	1.70E+11	3.40E+11
Kr-87	3.79E+10	7.59E+10	Xe-135m	9.53E+10	1.91E+11
Kr-88	4.49E+10	8.98E+10	Xe-138	1.17E+11	2.34E+11
La-140	5.51E+05	1.10E+06	Y-91m	1.68E+04	3.37E+04
Mn-54	2.95E+04	5.90E+04	Y-92	3.70E+05	7.40E+05
Mn-56	1.68E+05	3.35E+05	Zn-65	1.43E+05	2.86E+05
Mo-99	9.56E+02	1.91E+03	Zr-95	6.76E+03	1.35E+04
Na-24	1.06E+05	2.13E+05	Zr-97	1.23E+04	2.46E+04

The annual discharge limits proposed for significant radionuclides in aqueous and gaseous effluents discharged from the RR SMR are shown in Table.29.6-3.

Table.29.6-3. Proposed Annual Limits for Radioactive Discharges for Significant Radionuclides

Discharge phase	Radionuclide/ radionuclide group	Predicted annual discharges (Bq/y)	Proposed annual (12 month rolling average) Limit (Bq)
Aqueous	H-3	8.88E+10	1.78E+11
	Cs-137	1.12E+08	2.24E+08

Gaseous	H-3	4.08E+10	8.16E+10
	C-14	1.80E+11	3.59E+11
	I-131	6.76E+07	1.35E+08
	Xe-133	6.22E+11	1.24E+12

The proposed annual limits are likely to represent a restrictive envelope for operational discharges from the RR SMR due to the conservative approach used to derive the headroom factor.

The predicted annual discharges will be revised using the definitive PST and better-defined performance criteria and design parameters for relevant RR SMR discharge abatement systems are available.

The derivation of headroom factors and proposed limits will also be reviewed when the PST and performance of treatment and discharge systems can be assessed with greater confidence, and consideration given to use of approaches other than the WCPD model used above. Any change to the approach for determining headroom factors in proposed limits from that outlined above will be reviewed and justified based on the information used.

The interim PST is deemed to include the expected events contribution, as detailed in Section 29.3 as it is not possible at this stage to disaggregate normal operations and expected events from the data. **{REDACTED FOR PUBLICATION}** the fault schedule will be reviewed as the design of the RR SMR progresses. This review will ensure that all expected events meeting the specified frequency criteria **{REDACTED FOR PUBLICATION}** are identified and included in discharge calculations. **{REDACTED FOR PUBLICATION}**

29.7 Comparison with Discharges from Similar Power Stations

29.7.1 Overview

The UK Discharge Strategy [3] states that "discharges from new nuclear power stations in England and Wales will not exceed those of comparable power stations worldwide ".To gauge the performance of the RR SMR design against comparable nuclear power stations, the predicted discharges of radioactive effluents from the RR SMR (Section 29.6.3) have been normalised (to 1GWe) and compared to normalised discharges from selected PWRs.

For this interim assessment, it was considered proportionate to focus the comparison to predicted discharges from PWRs designs previously submitted for GDA (UK EPR™, UK AP1000® and UK HPR1000) and Sizewell B, the only operational PWR in the UK. The UK EPR™, UK AP1000® and UK HPR1000 designs were also required to demonstrate during the GDA process that their discharges would be comparable to or lower than similar nuclear power stations worldwide, in compliance with the UK Discharges Strategy. It is reasonable to assume therefore, that if the RR SMR discharges are comparable with these designs, they will also meet the conditions of the UK Discharge Strategy

It is recognised that whilst the RR SMR design is based on the well-established PWR technology, there are certain aspects of the design (e.g., the primary coolant chemistry) that have similarity with other reactor designs. Thus, future iteration of this analysis will compare normalised RR SMR discharges to that from other reactor designs such as VVERs, BWRs and other SMRs that may have common features with the RR SMR (subject to data availability).

{REDACTED FOR PUBLICATION}

29.7.2 Comparison with other nuclear power stations

Table 29.7-1 and Table 29.7-2 compare the normalised annual discharges of significant aqueous and gaseous radionuclides (respectively) predicted to arise from the RR SMR to those reported in the GDA submissions for the UK EPR™ [41], AP1000® [42], and UK HPR [43], as well as the 5-year average of reported discharges from Sizewell B, taken from published RIFE Reports [44], [45], [46], [47], [35].

Table 29.7-1. Comparison of Normalised Annual Discharge of Gaseous Effluent (Bq/GWe)

Reactor	H-3	C-14	I-131	Noble gases	Other radionuclides*
RR SMR	8.63E+10	3.81E+11	1.43E+08	2.50E+12	2.18E+07**
UK HPR1000	7.07E+11	3.17E+11	1.48E+07	1.03E+12	3.27E+06
UK AP1000®	1.61E+12	5.42E+11	1.85E+08	ND#	1.20E+07
UK EPR™	2.9E+11	2.00E+11	2.90E+07	4.60E+11	2.30E+06
Sizewell B	5.21E+11	2.50E+11	1.40E+07	2.59E+12	4.84E+06

*Described as particulate beta for Sizewell B
**Data for RR SMR excludes the contribution from other radionuclides in the table
ND – No data

Table 29.7-2. Comparison of Normalised Annual Discharge of Aqueous Effluent (Bq/GWe)

Reactor	H-3	Cs-137	Other	C-14	Radioiodine
RR SMR	1.88E+11	2.37E+08	1.14E+09	1.07E+08	1.03E+07
UK HPR1000	2.28E+13	1.28E+10	2.82E+08	ND	ND
UK AP1000®	3.0E+13	ND	2.40E+09	3.00E+09	1.34E+07
UK EPR™	3.0E+13	ND	3.50E+08	1.30E+10	4.00E+06
Sizewell B	2.17E+13	4.05E+08	5.09E+09	ND	ND

As can be seen in the data presented in Tables, the predicted annual discharges of radionuclides in aqueous and gaseous effluent from the RR SMR are broadly consistent with the annual normalised discharges from the PWRs against which they are compared.

The following observations on RR SMR discharges relative to the comparator PWRs were made:

1. The predicted discharge of gaseous tritium is around an order of magnitude less than that predicted to be discharged by the other PWRs.

2. Discharges of gaseous carbon-14, iodine-131 and noble gases are slightly above the average but below the maximum discharge values for the other PWRs.

3. Discharges of other radionuclides (the sum of all other radionuclides for which discharge has been determined, except those listed in Table) in gaseous form are greater than the upper value for the other PWRs.

4. For aqueous discharges, predicted annual releases of all the radionuclide categories assessed, except radioiodine, are less than the average values from other PWRs

5. Aqueous radioiodine discharges for RR SMR are greater than the average of the comparator PWRs but less than the upper value reported for the other PWRs.

The lower tritium discharges reflect the boron-free chemistry of the RR SMR. The discharges of carbon-14, radioiodine, caesium-137, noble gases sit within the existing range of comparator PWRs.

The 'other radionuclides' annual discharge is slightly higher than those from comparator plants. Direct comparison of individual radionuclides within this category (for example, gaseous effluent in Table 29.4-18 with gaseous discharge data estimates at a similar stage in the project (e.g. Reference [48]) may show orders of magnitude difference in individual radionuclides. This is likely an assumption of both the generalised nature of the RR SMR Interim PST, compared to other designs, based on existing plant and have better defined PST, and an artefact of the cautious assumptions used for abatement plant where design information is not yet confirmed.

It is anticipated that the Interim PST will set an upper bound on discharges from the RR SMR. Future discharge estimates will be lower than those derived for this interim stage, as information on the definitive PST, abatement and discharge systems become available.

To enable a comparison of the performance of the RR SMR with a wider range of nuclear power stations, future iterations of this analysis will also be extended to include detailed data from global PWRs obtained from open sources such as:

1. European Commission Radioactive Discharges Database (RADD) [50], supplemented with nuclear power stations specification and performance data from the International Atomic Energy Agency's Power Reactor Information System (PRIS) [51]

2. National reports such as the US Nuclear Regulatory Commission's Radioactive Effluent and Environmental Reports [52] and the UK RIFE Reports (e.g Reference [36].

29.8 Conclusions

29.8.1 Conclusions

A preliminary assessment of potential discharges of aqueous and gaseous radioactive effluent from the RR SMR plant to the environment under normal operating conditions, has been carried out. The assessment was performed using the available RR SMR design information at PCD, reflecting the current level of maturity of the RR SMR, and published OPEX data from PWR plants. An account of the methods adopted, including underlying assumptions and parameter values used to perform assessments is detailed throughout the Chapter.

The best estimates of annual discharges were used to derive propose annual discharge limits for significant radionuclides, as required by regulatory guidance [4]. The predicted annual discharge limits were normalised to power output in order to compare against discharges from similar nuclear power stations. Aqueous radioactive effluent discharges were mostly found to be below the average of reported or forecasted discharges from comparable PWRs. The normalised annual discharge of gaseous radioactive effluent from the RR SMR were mostly consistent with the average of forecasted or reported discharges from other PWR plants, with the exception of discharges of 'other radionuclides'. The higher values observed for this group is likely an artefact of the cautious assumptions used to estimate the discharges from the RR SMR.

The Environment Agency's initial radiological assessment tool (IRAT) [52], [53] enables a conservative dose assessment of the impact of discharges on members of the public and the environment. Results of the initial assessment of the impact of RR SMR radioactive effluent discharges to the RR SMR Generic Site described in Chapter 2 of the E3S case indicated that the total dose to a representative member of the public from all relevant exposure pathways is 0.012mSv/y. This is well below the source dose constraint of 0.3mSv/y [54] and the more restrictive constraint of 0.15mSv/y recommended to be applied at the design stage of new nuclear facilities by the Health Protection Agency (now subsumed into the UK Health Security Agency) [55]. The dose rate to the worst affected organism was estimated at 0.02μGy/h, which is a few orders of magnitude less than the statutory guideline value of 40μGy/h [6]. Details of the initial assessment of radiological impacts due to predicted discharges of radioactive effluent from the RR SMR are presented in Chapter 30.

The interim calculations and assessments will be re-evaluated as the design of the RR SMR progresses. **{REDACTED FOR PUBLICATION}**

29.9 Assumptions & Commitments on Future Permit-holders/Dutyholders

29.9.1 Assumptions and Commitments on Future Permit Holders /Dutyholders

The intention is that the E3S Case will capture assumptions and commitments for future permitholders/dutyholders/licencees. Environmental assumptions and commitments for permitholders have not been formally captured at this stage, but will be included in future revisions of the GER.

29.10 Forward Action Plan

29.10.1 Forward Action Plan

{REDACTED FOR PUBLICATION}

{REDACTED FOR PUBLICATION} Table 29.10-1. Forward action plan for quantification of discharges from RR SMR

29.11 References

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29.12 Acronyms and Abbreviations

AP	Activation Products
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
ANSI/ANS	American National Standards Institute/American National Standards
ActP	Actinide Products
BAT	Best Available Techniques
BE	Best Estimate
BSS	Basic Safety Standards
BWR	Boiling Water Reactor
CA	Cycle Average
CARS [MAJ]	Condenser Air Removal System
CDS [KTA]	Collection and Drainage System
CMA	Chemistry Monitoring and Assessment
CP	Corrosion products
CVCS [KB]	Chemistry and Volume Control System
DB	Design Basis
DECC	Department for Energy and Climate Change
DFs	Decontamination Factors
DOORS	Dynamic Object Oriented Requirements System
E3S	Environment, Safety, Security and Safeguards
EA	Environment Agency
EC	European Commission
EPR16	Environmental Permitting (England and Wales) Regulations 2016
EPRI	Electric Power Research Institute
FAB40	Upender Pit
FAE	Refuelling Cavity
FAF	Refuelling Pool

FBC	Ultrasonic cleaning station
FCK	Fuel Transfer Channel
FCST	Fuel Crud Source Term
FP	Fission products
FPSS [FAT]	Fuel Pool Supply System
GDA	Generic Design Assessment
GDPs	Generic Developed Principles
GER	Generic Environment Report
GWe	Gigawatts electrical
GWTS [KPL]	Gaseous Waste Treatment System
HEPA	High-Efficiency Particulate Air (filter)
HVAC [KL-]	Heating, Ventilation and Air Conditioning
ICRP	International Commission on Radiation Protection
IRAT	Initial Radiological Assessment Tool
IX	Ion Exchange
KB-	Primary Coolant Treatment System
KNF10	Processing and Treatment System for Primary Liquid Effluent
KNF20	Processing and Treatment System for Spent Liquid Effluent
LWR	Light Water Reactors
LMDS [KNF30]	Liquid Effluent Monitoring and Discharge System
MAB	Mass and Activity Balance
MPa	Megapascal
mSv	Millisievert
mSv y^{-1}	Millisieverts per year
MWe	Megawatts electrical
MWt	Megawatts thermal

NRW	Natural Resources Wales
ONR	Office for Nuclear Regulation
OPEX	Operating Experience
OSPAR	Oslo-Paris Convention (Convention for the Protection of the Marine Environment)
PC	Partition Coefficients
PCD	Preliminary Concept Definition
PSST	Primary System Source Term
PST	Primary Source Term
PWR	Pressurised Water Reactor
R01	Reactor Island
RCDT	Reactor Coolant Drain Tank.
RCS	Reactor Coolant System
RGP	Relevant Good Practice
RIFE	Radioactivity In Food and the Environment
RO	Reverse osmosis
ROPs	RSR Objective and Principles
RP	Requesting Party
RPV	Reactor Pressure Vessel
RR SMR	Rolls-Royce Small Modular Reactor
RSR	Radioactive Substances Regulations
RWMA	Radioactive Waste Management Arrangments
SFAIP	So Far As Is Reasonably Practicable
SFP [FAB10]	Spent Fuel Pool
SFPCS [FAK]	Spent Fuel Pool Clean-up System
SG	Steam Generator
SGBD	Steam Generator Blowdown
SMR	Small Modular Reactor
SSCs	Structures, Systems and Components

SSST	Secondary System Source Term
T01	Turbine Island
TDS	Total Dissolved Solids
TOC	Total Organic Carbon
UK	United Kingdom
UST	User Source Term
VRFs	Volume Reduction Factors
VVER	Water-Water Energetic Reactor
WCPD	Worst Case Plant Discharge
WQS	Water Quality Specification
WWTP [GM-]	Wastewater Treatment Plant
$\mu Sv\ y^{-1}$	Microsieverts per year
$\mu Gy\ h^{-1}$	Micrograys per hour