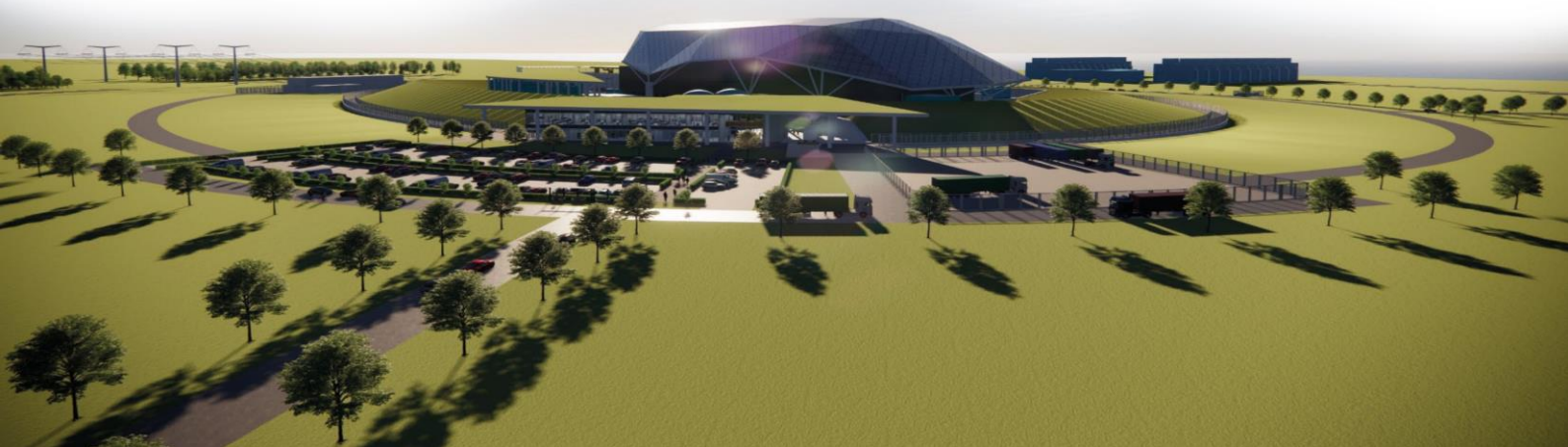




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# **Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 29: Quantification of Radioactive Effluent Discharges and Proposed Limits**



## Record of Change

Date	Revision Number	Status	Reason for Change
30/03/2023	1	Issue	Formal version for issue
31-3/2023	2	Issue	Final version for issue incorporating formatting changes for E3S consistency (full document)
31/05/2024	3	Issue	<p>Updated to correct revision history status at Issue 2.            Chapter changes include:            Updated to incorporate outcomes from technical review.,</p> <ul style="list-style-type: none"> <li>• Improved signposting to other E3S chapters and technical notes with summaries of contents</li> <li>• Revised estimates of predicted liquid and gaseous effluent discharges based on updated primary source terms and primary systems source terms, and RR SMR systems design at RD7/DRP1 maturity (section 29.2)</li> <li>• Revised estimation of predicted monthly discharges and annual permit limits based on the revised discharges and OPEX from similar nuclear power plant (sections 29.3 and 29.4)</li> <li>• Comparison of predicted discharges from the RR SMR with those from similar nuclear power plant.</li> </ul>

## Executive Summary

Environment, Safety, Security and Safeguards (E3S) Case Version 2, Tier 1, Chapter 29: Quantification of Radioactive Effluent Discharges and Prospective Permit Limits 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 Rolls-Royce small modular reactor (RR SMR) at reference design 7 (RD7)/design reference point 1 (DRP1), 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 headroom factors derived from variability in PWR operating experience (OPEX) is provided.

The annual discharges of aqueous radioactive effluent predicted to arise from the RR SMR, normalised to 1 GW<sub>e</sub>, are mostly found to be below the average of reported or forecasted discharges from comparable PWRs. Where normalised discharges are higher than reported values, it is considered likely to be an artefact of the conservative assumptions used in determining RR SMR discharge estimates and will be investigated further.

The design of the RR SMR is still under development and the source term will continue to be reviewed and updated as the design develops. The discharge calculations presented in this chapter will be further updated as underlying assumptions, parameter values, operating experience data and engineering design are reviewed. Forward actions (FAs) included in this chapter indicate the key revisions and updates to be undertaken prior to the next version of the E3S case.

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## 29.0 Introduction to Chapter

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### 29.0.1 Introduction

Chapter 29 of the Rolls-Royce Small Modular Reactor (RR SMR) Environment, Safety, Security and Safeguards (E3S) Case is a Tier 1 chapter as defined in E3S Case Version 2, Tier 1, Chapter 1: Introduction [1].

The chapter 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 at RD7/DRP1 and will be updated using the information that becomes available as the design matures.

The information presented in this chapter provides quantitative estimates of, and proposed limits for, aqueous and gaseous radioactive wastes discharged to the environment, and compares the estimated discharges with those from similar nuclear power stations across the world.

### 29.0.2 Scope and Maturity

This chapter provides an outline of progress made towards quantifying the environmental discharges based on the design of the RR SMR at RD7/DRP1 and highlights the forward actions (FAs) which will be undertaken to fully achieve the objectives for the mature RR SMR design. The chapter outlines the methodologies used and the conservative assumptions applied where RR SMR-specific data is not yet available. As the design of the RR SMR progresses it is anticipated that confidence in the estimated discharges will improve.

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.

The 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 29.1.4. 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 (including liquid waste not suitable for discharge to the environment) is outside the scope of this chapter and is addressed in an integrated waste strategy (IWS) [2], summarised in E3S Case Version 2, Tier 1, Chapter 11: Management of Radioactive Waste [3], E3S Case Version 2, Tier 1, Chapter 12: Radiation Protection [4] and E3S Case Version 2, Tier 1, Chapter 20: Chemistry [5].

Structures, systems and components (SSCs) from Reactor Island (RI) [R01] and Turbine Island (TI) [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 E3S Case Version 2, Tier 1, Chapter 25: Detailed Information About the Design [6].

This chapter draws upon the information presented in other chapters of the RR SMR E3S Case. For example, E3S Case Version 2, Tier 1, Chapter 1: Introduction [1] provides an introduction to the overall case, and E3S Case Version 2, Tier 1, Chapter 25 Detailed Information About the Design [6] provides signposting information to further chapters which include detailed information of plants, systems and processes, which have a bearing on radioactive waste (solid, liquid and gaseous) generation, treatment, measurement, assessment and disposal.

### 29.0.3 Claims, Arguments and Evidence Route Map

The information presented in this chapter provides some of the evidence supporting the fundamental claims for quantification of discharges and prospective permitting limits. Further detail on the claims for environmental aspects of the E3S case can be found in E3S Case Version 2, Tier 1, Chapter 27: Implementation of Best Available Techniques [7].

The top-level claim applicable to this chapter is:

**Claim 29:** SMR gaseous and aqueous waste arisings during normal operation are well understood and defined, with discharges optimised using BAT and minimised to ALARA<sup>1</sup>.

As noted in section 29.0.2 on scope and maturity, and in Appendix A (section 29.8), waste arisings in respect of solid waste, liquid effluent unsuitable for disposal to the environment, and activated structures are addressed in other E3S Case Version 2, Tier 1, Chapters.

A decomposition of this claim into level 1 sub-claims, arguments, and link to the relevant sections of this report is provided in Appendix A (section 29.8), and these claims will be further broken down and more detailed evidence presented in future revisions of this chapter. The complete suite of evidence to underpin the claims in the E3S case will be generated through the E3S case programme and documented in the claims, arguments, evidence (CAE) route map [8], described further in E3S Case Version 2, Tier 1, Chapter 1: Introduction [1].

The three main sub-claims which are further addressed in this chapter include:

- Quantitative estimates of discharges and disposals of aqueous and gaseous effluents are defined, bounded by the RR SMR normal operations source term.
- Discharges of aqueous and gaseous radioactive effluent to the environment from the RR SMR are reduced to levels that are as low as reasonably achievable (ALARA).
- Radioactive effluent discharges from the RR SMR, corrected for differences in design, are comparable to discharges from similar nuclear power stations globally.

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<sup>1</sup> Different regulatory regimes in the field of radiological protection in the UK use different terminology and have their own guidance, including reducing risks to as low as reasonably practicable (ALARP) and reducing exposure to ionising radiation to as low as reasonably achievable (ALARA). The terminology is broadly synonymous, with both ALARA and ALARP incorporating considerations on economic, environmental and societal factors.

This chapter quantifies radioactive effluent discharges from all modes of normal operation, during the operational lifetime of the RR SMR and provides an overview of the proposed discharge limits, as defined at RD7/DRP1 level of design maturity.

## 29.0.4 Applicable Regulations, Codes and Standards

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) [9] set out the principles and fundamentals for radiation protection, which are adopted worldwide. The European Commission (EC) Directive 2013/59/EURATOM [10] applies these principles to a system of radiation protection and control, to be transposed into the domestic legislation of the European Union (EU) Member States<sup>2</sup> Of particular relevance to this chapter are the requirements for States to keep doses to members of the public 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.

### *Regulation in England and Wales*

Discharges of radioactive substances from nuclear power stations to the environment are controlled under the Radioactive Substances Regulation (RSR), set out in Schedule 23 of the Environmental Permitting (England and Wales) Regulations 2016 (as amended) (EPR16) [11]. The regulation ensures 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 best available techniques (BAT) is 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 as low as reasonably achievable (ALARA).

EPR16 [11] requires regulators to, in the exercise of their function, ensure that:

- 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 ALARA, taking into account economic and social factors.
- The sum of the doses arising from such exposures does not exceed the individual public dose limit of 1 mSv per year.
- The dose to an individual due to discharges from any source (since 13th May 2000) does not exceed 0.3 mSv per year.
- The dose to an individual due to discharges from any single site does not exceed 0.5 mSv per year.

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<sup>2</sup> 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.



- 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)<sup>3</sup> issued Statutory Guidance to the Environment Agency (EA)<sup>4</sup> concerning the regulation of radioactive discharges to the environment [12], 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 10 µSv per year, below which the EA should not seek to further reduce the discharge limits that are in place, provided that the holder of the permit continues to apply BAT.

The Statutory Guidance issued by DECC [12] 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 the requirements of the Oslo-Paris Convention for the Protection of the Marine Environment (OSPAR) for a progressive reduction in radioactive discharges, whilst not compromising UK domestic energy policy<sup>5</sup>, and is captured in the generic design assessment (GDA) requirements.

**Alignment with the Regulatory Objectives and Principles**

The EA’s RSR objective and principles (ROPs) [13] set out the regulatory principles the EA 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 EA’s expectations on permit holders carrying out radioactive substances activities [14]. Reference [15] indicates how the RR SMR E3S case will incorporate the GDPs throughout the document identifying the key principles for each relevant chapter of the E3S case. The key GDPs directly relevant to this chapter are identified in Table 29.0-1.

**Table 29.0-1 Relevant RSR Generic Principles**

GDP ID	Title	Principle
Radioactive substances management developed	Radioactive Substances Strategy	A strategy should be produced for the management of all radioactive substances.

<sup>3</sup> Subsumed into the Department for Business, Energy & Industrial Strategy (BEIS) in July 2016.

<sup>4</sup> Since 2013, regulation of radioactive substances in Wales has been the responsibility of Natural Resources Wales

<sup>5</sup> 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.

GDP ID	Title	Principle
principle, RSM DP 1		
RSM DP3	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.
RSM DP12	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.
Radiological protection developed principle, 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
Engineering developed principle, 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 GDPs, in particular those listed above, will be taken into account in meeting the objectives for quantification of discharges to the environment in this chapter.

## 29.1 Source Term Development

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### 29.1.1 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 [5], [16]. 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:

- Present in the coolant and entrained gases
- Fixed or deposited onto surfaces of structures such as reactor core and primary circuit pipework
- Accumulated in abatement structures such as filters and demineralisers.

The RR SMR source term also includes source terms associated with spent fuel and activated structures; however, these are out of the scope of this chapter.

The purpose of generating the RR SMR source term is to quantify the radioactive inventory of the RR SMR so that the radioactive hazard in the reactor, supporting systems and circuits can be understood. The source term is used to support the development of the RR SMR design and the E3S case to demonstrate that risks relating to radioactivity have been reduced to levels that are ALARP and that BAT has been applied, so resulting discharges are ALARA.

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, focusing on the source term in the reactor core (the primary source term (PST)). The PST is used as a basis for deriving the radioactive inventories in supporting plant systems in the primary circuit (primary system source term (PSST)) and secondary circuit (secondary system source term (SSST)), which are used in the estimation of radioactive effluent discharges to the environment. Further information on the source term types is provided in E3S Case Tier 1 Chapter 20: Chemistry [5].

#### **Source Term Value Categories**

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. The source term value categories being developed for the RR SMR are described in E3S Case Version 2, Tier 1, Chapter 20: Chemistry [5]. In line with relevant good practice (RGP) radioactive effluent discharge estimates are derived using the best estimate (BE) source term values, such that the discharge estimates are representative of realistic normal operations conditions

(reference [17], and previous assessments, e.g. [18] [19]) representative condition that is realistic so as not to result in over-specification of the source term for plant systems.

The BE normal operations source term, including the PST and PSSTs described in Reference [20] is used throughout this chapter as the basis for the quantification of radioactive discharges to the environment.

## 29.1.2 Radionuclide Production Mechanism

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

- Fission products (FP)
- Actinide products (ActP)
- Corrosion products (CP)
- Activation products (AP).

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 FP and a CP, 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 [21].

An overview of the generation of these radionuclides in the reactor is summarised in E3S Case Version 2, Tier 1, Chapter 20: Chemistry [5].

## 29.1.3 Source Term Derivation Methodology

Details on the strategy for the derivation of the normal operation source term for the RR SMR is provided in the Normal Operation Source Term Strategy Report [16]. This report defines what is required to derive the normal operation source term, covering its scope and general principles for derivation and justification. It also covers the main interfaces with the normal operation source term, in addition to how the normal operation source term is developed, structured and delivered. The normal operation source term document structure is also described in the strategy report; it shows how the source term will be produced in a structured and staged approach with three tiers of information. The radionuclide list (and related methodology) associated with the normal operation source term for the RR SMR is documented in the Normal Operation Source Term Radionuclide Selection Report [21]. The development of this list takes account of RGP in the form of codes and standards, and datasets from similar technologies, and considers design choices specific to the RR SMR. It outlines the rationale and technical basis for why radionuclides are included in the normal operation source term.

Datasets for each of the constituent source terms of the normal operation source term are recorded as live datasets in the RR SMR requirements database (DOORS), as separate modules [20]. The derivation of the PST, PSST and SSSTs are described in E3S Case Version 2, Tier 1, Chapter 20: Chemistry [5].

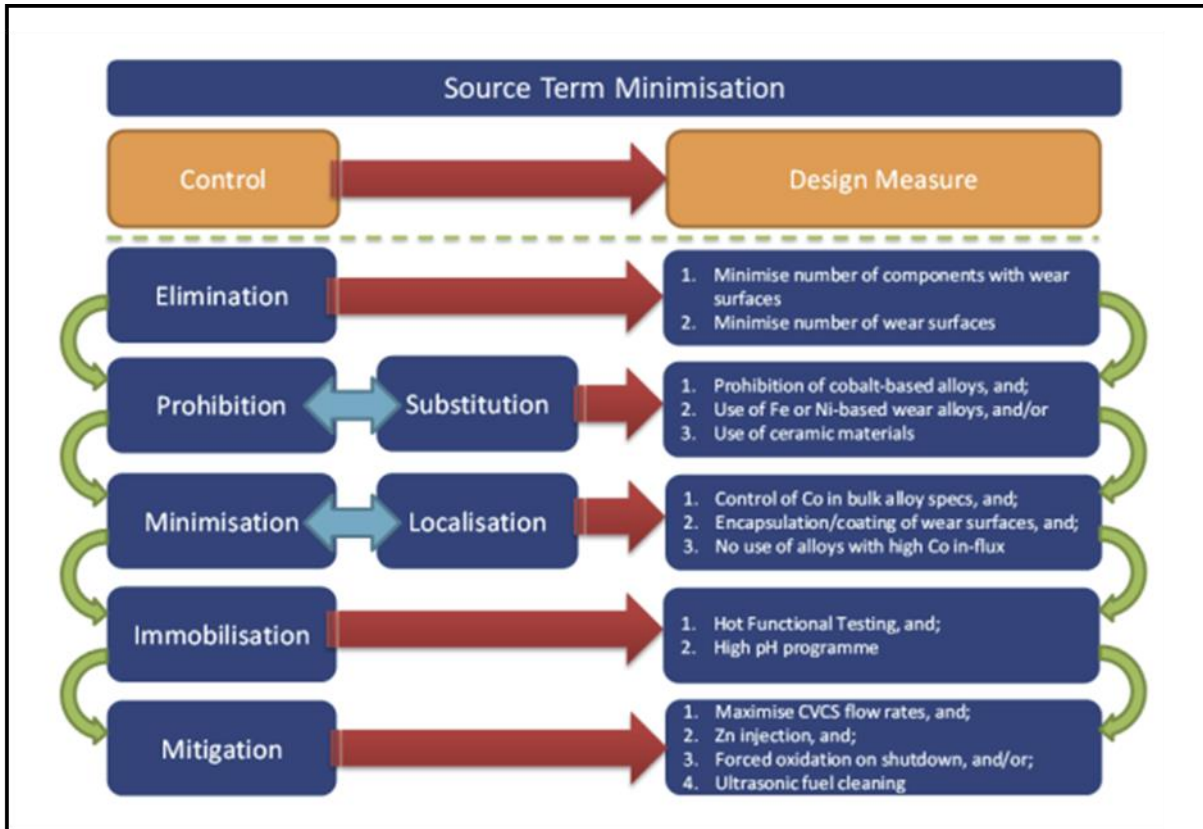
### ***Source Term Reduction/Optimisation Philosophy***

The development of the RR SMR from fundamental PWR technology has afforded the opportunity to identify and implement RGP and source term reduction measures into the fundamental design of the RR SMR. This ensures disposals are ALARA 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 substances 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 ALARP and to use BAT [22]. A hierarchy of controls has therefore been conceived, the application of which will, and can be demonstrated to ensure optimisation of the source term [23], in-line with the RR SMR E3S principles [22], as illustrated in Figure 29.1-1, which used cobalt-60 source term reduction as an example.

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. 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 ALARA and keeping discharges as low as reasonably practicable (ALARP). The PST will account for the opportunities for reduction and minimisation of the source term through plant design, plant chemistry and choice of fuel specification. Details of the mandatory requirements and recommendations applied to the design are recorded in the RR SMR requirements database (DOORS).



**Figure 29.1-1: Hierarchy of Source Term Controls for Co-60**

**Operating Phases**

Six operating modes, characterised by RCS temperature, pressure and refuelling status, are defined in the RR SMR operating philosophy, outlined in E3S Case Version 2, Tier 1, Chapter 13: Conduct of Operations [24]. 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:

- Start-up
- Power operation
- Shutdown
- Refuelling outage.

The conditions under which each operating phase applies is described in Reference [16].

A cautious estimate of an 18-day outage period is assumed, to provide some margin in estimated discharges during the shutdown phase. These estimates will be kept under review [25].

Determination of the PST will take account of the changes in radionuclide concentration and distribution across the RR SMR during the different phases of operation.

## 29.1.4 Expected Events

Expected events (EEs) are described in EA guidance as “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”, “[...] expected to occur over the lifetime of the facility” [26]. For purposes of setting discharge limits under RSR, EEs are evaluated as a constituent of normal operations that need to be accounted for when quantifying radioactive waste arisings, to ensure that operators have sufficient flexibility to operate the plant and are not unduly constrained.

These EEs are considered equivalent to anticipated operational occurrences (AOOs) advanced by competent international organisations such as the Association of Regulators of Western Europe (WENRA) [27] and the International Atomic Energy Agency (IAEA) [28], and enshrined in the civil nuclear safety regimes of several leading nations for example References [29], [30] and [31] described in the Safety Analysis topic area.

The criterion for designating faults as AOOs is an initiating event frequency (IEF) of  $1.0E-02$  per year (pa) or greater. This criterion is consistent with the condition ‘expected to occur over the lifetime of the facility’ in EA guidance and fitting for the RR SMR, which has a design life of 60 years. The criterion of  $IEF \geq 1.0E-02$  was therefore deemed to provide a suitable basis for designating EEs for the RR SMR E3S case. This is consistent with plant states DBC-2i and DBC-2ii IDs, which cover design basis (DB) conditions (normal operation: abnormal conditions) and design basis conditions (fault conditions) within the  $>1E-02$  pa frequency band described within the RR SMR E3S Design Principles document [22].

An initial shortlist of credible EEs for the RR SMR E3S Case was derived from published literature, RR SMR stakeholder engagement, and information contained in design documentation produced by the RR SMR safety and chemistry teams [32]. The list of faults falling within scope of the expected events criteria is kept under review as the RR SMR plant design progresses and the RR SMR safety analysis documentation is developed. Reference [33] describes the review of EEs based on fault studies documentation available at RD7/DRP1 [34] [35]. The main expected events identified as being relevant to the RR SMR normal operation source at RD7/DRP1 are described in Reference [33] and summarised below:

- Fuel pin failure. Fuel failures may lead to the release of volatile FPs into the primary coolant and could make a significant contribution to normal operations source term. 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. Initial estimates of fuel failure rates based on OPEX (references [36], [37]) will be refined as core design progresses and information from the fuel supplier becomes available.
- Primary-to-secondary circuit SG tube leak. This event is expected to result in the transfer of radioactivity from the RR SMR primary to secondary circuit as a result of degradation of the SG tubes caused by various failure mechanisms. A SSST covering 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. Factors such as materials used for construction of SG tubes and coolant chemistry will be chosen to minimise failure rates.
- Unscheduled/automatic reactor shutdown. Several events identified in the fault schedule may result in an unscheduled shutdown of the reactor and subsequent enhanced release of FP



and CP into the primary coolant. The associated event frequencies are summated to determine the overall predicted frequency of unscheduled shutdowns. 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.

The list of RR SMR expected events shall be kept under review and periodically updated using information captured in the Fault Schedule at key design reference points.

The approach for estimating the contribution of EEs 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 (for example, using correction factors) to account for EEs. Considering the diversity of approaches that will be used to generate the source term, it is likely that both approaches for estimating the contribution to radioactive discharges to the environment will be adopted.

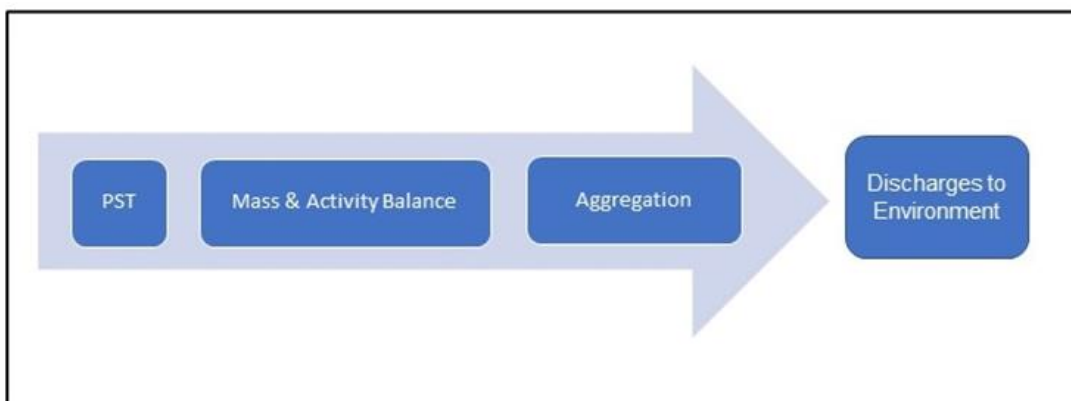
**FA 29.1** The Fault Schedule shall be reviewed periodically to ensure all expected events are accounted for, and, where possible, their contribution is quantified.



## 29.2 Quantification of Discharges to the Environment

### 29.2.1 Overview of Calculational Approach

To calculate the quantity of radioactive materials present in effluents at various points and stages of the RR SMR fuel cycle, radioactive effluent transfer and removal factors (derived from empirical observations from operating PWRs and engineering judgements published in peer-reviewed reports) are applied to the PST and the design parameters (effluent masses and flow rates) in mass-activity balance (MAB) models. The overall approach is illustrated in Figure 29.2-1.



**Figure 29.2-1: Overview of the Approach for Estimating Effluent Discharges to the Environment**

Discharges of radioactive effluent to the environment are estimated using a MAB model to calculate the activity exiting the various liquid and gaseous radioactive effluent treatment systems.

The MAB approach also applies a decontamination factor (DF) and volume reduction factor (VRF) to represent the partitioning of radioactivity from the liquid and gaseous throughput to the solid/ semi-solid phase of the abatement system. The DFs and VRFs 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 Gaseous Radioactive Effluent Treatment System (GRETS) [KPL] where radioactive decay is the operating principle, and for radionuclides with short half-lives where decay is an important mechanism.

In the case of gaseous discharges, the total gaseous radioactive effluent discharges to the environment are derived from the aggregation of the various treated gaseous effluent streams identified in section 29.2.3 to derive the total discharges from the plant to the environment.

All treated radioactive liquid effluent streams are transferred to the liquid radioactive effluent processing system [KNF30] for storage and recycling, or discharge to the environment, and thus an aggregation step is not required within the calculation.

## 29.2.2 Sources and Estimates of Aqueous Radioactive Effluent Discharges

### *Sources of Aqueous Effluent*

Liquid radioactive effluents are categorised and segregated based on their source and expected levels of contamination. Two primary sources of liquid radioactive effluents have been characterised for the RR SMR [38].

- 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.
- Spent liquid effluent comprises effluent streams, originating from different sources (process, chemical and floor drains), collected for treatment by the sumps and vessels of the Collection and Drainage System (CDS) [KTA].

Further details on the estimation of the liquid radioactive effluent volumes are presented in Reference [39], which indicates that a significant proportion of the effluents will be recycled, with the cautious assumption of a 1 off 74 m<sup>3</sup> tank being discharged per cycle.

In addition, there is potential for SG feedwater or secondary coolant to become contaminated, for example due to tube leakage, and thus the blowdown will not be suitable for treatment at the wastewater treatment plant (WWTP) [GM-]. The approach for handling contaminated SG blow down (SGBD) depends on the degree of leakage and level of radioactive contamination of the secondary coolant (or any set operating limits). It is anticipated SGBD with limited contamination from small tube leaks will be transferred to the KNF system [KNF30] for discharge, whilst SG blowdown that is more heavily contaminated from tube rupture is transferred to [KNF20] for treatment and then [KNF30] for discharge. Further detail of the [KNF] system is provided below.

### *Processing and Treatment of Liquid Effluent*

The [KNF] system collects the liquid effluents 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 processes, including use as make-up demineralised water, or otherwise renders the effluent suitable for discharge to the environment.

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

- Processing & treatment system for primary liquid effluent ([KNF10])
- Processing & treatment system for spent liquid effluent ([KNF20])
- Liquid effluent monitoring and discharge system ([KNF30]).



Detailed information on the [KNF] subsystems is presented in E3S Case Version 2 Tier 1 Chapter 11: Management of Radioactive Waste [3]. The key processes and effluent routes are highlighted in the process flow diagram in Figure 29.2-2.

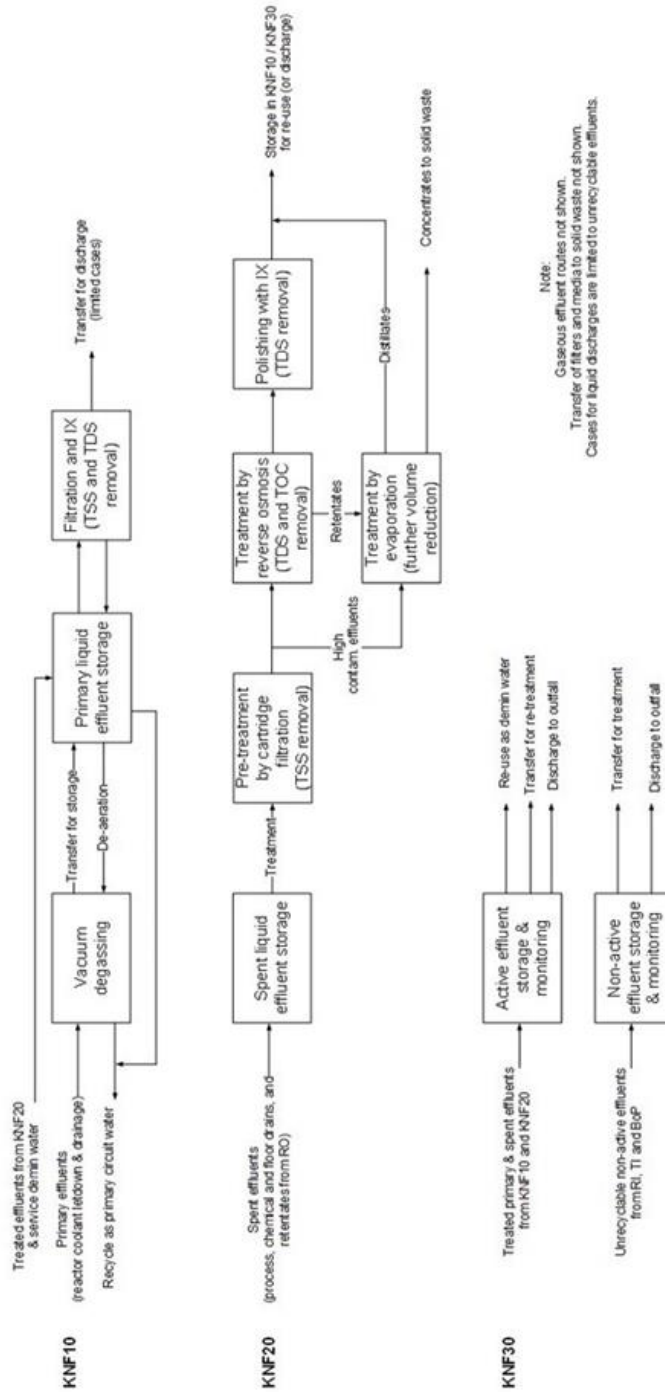


Figure 29.2-2: Process Block Flow Diagram for Liquid Radioactive Effluent Processing System [KNF]

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 RI. This is in accordance with the corporate objective of designing a product that can be operated with minimal discharge of liquid radioactive effluent to the environment.

The primary reason for discharge of treated liquid effluents is for the purposes of managing tritium activity. As part of normal operations, it is expected that a tritium bleed may be required as part of selected cycles, where there may be increased tritium accumulation (as discussed in reference [38]) to manage tritium concentration in the primary circuit. Bleed of treated, tritiated effluent will be coupled with makeup of the primary circuit with non-tritiated, reactor grade water to reduce tritium concentration in the primary circuit. This may be required to meet reactor pressure vessel (RPV) head-lift criteria as part of refuelling operations, to ensure safe access for operators during outage. The RPV head-lift criteria is still to be determined for the RR SMR and will be dependent on dose assessments for refuelling outages and is noted as an action for further work [40]. There may be other scenarios which necessitate liquid discharges, such as for water balance purposes, but this is expected to be infrequent and a general water balance assessment across RI is noted as an action for further work [40] to understand any potential requirement for water balance discharges.

The let-down coolant is degassed by the KNF10 vacuum degasser before being purified by the KNF20 system and discharged to the environment via the KNF30 system. The [KNF10] vacuum degasser is assumed to have a degasification factor of 50 [41], [42]. The level of decontamination of liquid throughput achieved by the RO unit, ion exchange (IX) beds and evaporator in [KNF20] are radionuclide-specific, based primarily on [43], and assumed performance factors are provided in References [40] and [42].

For the purposes of the liquid discharges MAB, gas fractions are assumed to be 1 for noble gases, 0.9 for carbon-14 [44] and 0.01 for halogen species [43] and are assumed to be unaffected by the treatment techniques. Tritium is also assumed to be unaffected by the treatment processes described.

The DFs applied to the [KNF] system at this stage of design are largely based on Reference [43]. These values will be updated with optimised, RR SMR-specific DFs and performance data obtained from treatment system manufacturers/vendors or commissioned studies.

### ***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 liquid effluent as no suitable cost-effective techniques for removing tritium from large volumes of liquid effluent have been identified for the RR SMR design at RD7/DRP1. Thus, disposals resulting from the requirement for a tritium bleed, as described above, are likely to be the primary source of effluent for discharge to the environment.

Treated effluents held in [KNF30] tanks would normally be reused as demineralised water within the RI systems. On the occasions when treated effluent in [KNF30] tanks cannot be reused or recycled within the RI and need to be discharged to the environment (for water balance purposes or to manage tritium activity), the effluent will be released to the environment via a single point. 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. A sampling and monitoring philosophy is under development to ensure the most appropriate locations for taking



representative samples are identified. 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] is determined from the MAB model. The inputs and assumptions for the liquid effluent processing MAB are described in Reference [40].

A discharge volume of 74 m<sup>3</sup> /cycle is currently assumed as part of the tritium bleed to manage tritium concentration in the primary circuit for a cycle where a tritium bleed is required (aligned to a [KNF30] tank volume's worth of effluent). This volume of coolant is assumed to be let down via the CVCS to [KNF10], degassed by the [KNF10] vacuum degasser before being purified by the [KNF20] system and discharged to the environment via the [KNF30] system. All other feed volumes to [KNF] are not considered, as drainage volumes are expected to be treated for recycle.

The source term input to the MAB is the concentration activity from the CVCS PSST. The processing (filtration and IX) in the CVCS results in a source term for [KNF] with reduced activity of particulate radionuclides and comparatively higher activity concentration of noble gases. Radionuclide activity concentrations are available for the CVCS PSST for shutdown and power operations phases. The source term input for liquid discharges is the CVCS PSST BE power operations source term, as defined at RD7/DRP1. This is deemed relevant for the tritium bleed scenario as it is assumed that the tritium bleed will occur during the power operations operating phase, via the CVCS system.

As described, the processing and management of 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, primarily as a result of tritium bleeds required for operator safety. Discharges of radioactive liquid effluent are therefore expected only on a relatively infrequent basis. 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.

The estimated annual activity of key radionuclides in the aqueous effluent discharge stream is presented in Table 29.2-3. A correction factor (0.67) is applied to the cycle discharge values determined in Reference [40] to estimate the activity generated in one year of operation. For the purposes of this chapter, it is assumed there will be an annual discharge equating to one year's worth of normal operations liquid effluent production (0.67 x activity generated in 18-month cycle)

### 29.2.3 Sources of Radioactive Gaseous Effluent Discharge

This section provides estimates of the gaseous radioactive effluent discharged from the RR SMR. The sources described below have been identified as routes from which gaseous radioactive effluent will be discharged during the course of normal operations, including from expected events as described in section 29.1.4. The list of sources will be kept under review as the design develops and the Fault Schedule is reevaluated. The total gaseous effluent discharge is derived from aggregation of the discharges from each of the individual sources.

#### ***The Processing and Treatment System for Gaseous Radioactive Effluent Treatment System [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 FPs (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 RCDT in the CDS [KTA] [45].

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 FP 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 (> 200 m<sup>3</sup> 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 FP effluent. Similarly, the RCDT in CDS 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<sup>6</sup> and auxiliary areas of the RI. This covers the containment building, auxiliary building and the area around the Spent Fuel Pool [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 [KPL]) via the gaseous emission exhaust stack [KLS].

### ***Air Removal and Evacuation System [MAJ]***

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 [MAJ] and discharged to the environment.

## **29.2.4 Discharges from the Gaseous Radioactive Effluent Treatment System [KPL]**

The [KPL] system consists of compressors delivering nitrogen cover gas to various systems within RI, a recombiner (for removal of hydrogen) and delay beds for abatement of FPs prior to discharge to the environment from the stack. The nitrogen cover gas purges radioactive gases (and hydrogen) as they degas from stored active coolant in tanks and vessels. Whilst hydrogen is removed through

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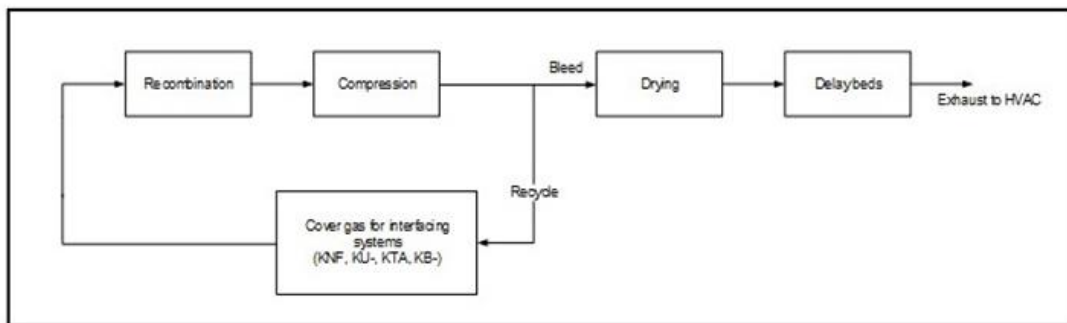
<sup>6</sup> Controlled areas, under Regulation 17(1) of the Ionising Radiation Regulations 2017 (The Stationery Office), are designated as such by the employer due to the need for special procedures to restrict exposure to ionising radiation, or where limits of 6 mSv/y effective dose, 15 mSv/y to the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

the recombiner, radioactive FP gases then circulate through the [KPL] cover gas network, until an increase in cover gas network pressure occurs. This has the effect of diverting radioactive gaseous effluent to the delay bed line, where activated charcoal beds provide hold-up and decay of xenon and krypton, prior to release of radioactive gaseous effluent to the stack via the [KLF] subsystem of the HVAC system. The delay line also contains a desiccant dryer, for further moisture removal in [KPL].

It is expected that the bulk of the activity released into [KPL] will result primarily from the [KNF] vacuum degasser operating at the shutdown transient (where the entire volume of primary coolant is degassed by the vacuum degasser to remove hydrogen and FP gases from the primary circuit in preparation for RPV head-lift for refuelling). However, the vacuum degasser also operates during power operations where it degasses any coolant which is letdown from the RCS for storage in [KNF].

During normal operation of [KPL], the net [KNF] volume remains largely unchanged, as letdown of coolant to [KNF] tanks is balanced by removal of makeup water from another [KNF] tank. In this scenario the nitrogen cover gas is recycled in a semi-closed loop, as overall volume in the [KNF] system remains the same. A proportion of the gas is bled to the delay beds as a result of surges in volumes in the system when there is a net increase in the volume of liquid in the [KNF10] tanks. The most pronounced case of this volume surge will be due to the thermal expansion/steam bubble formation in the primary circuit during reactor start up. Smaller volumes from thermal expansion transients (where coolant is letdown to [KNF10] without coupled make-up) may also contribute to discharges via [KPL].

A block flow diagram highlighting the key processes in managing gaseous radioactive effluent discharges is shown in Figure 29.2-3



**Figure 29.2-3: Block Flow Diagram for the Gaseous Radioactive Effluent Treatment System [KPL]**

The hold-up or delay times used in designing the KPL system are typical for PWR and Boiling Water Reactor (BWR) sizing as outlined in Reference [46] (40 hours for krypton, 40 days for xenon and radioiodine). Activation product gases are assumed to take 0.1 hours to pass through the KPL system, enabling short-lived radionuclides such as N-16 to decay away. It is assumed that argon-41 and carbon-14 are unaffected by the delay beds, other than the 0.1-hour decay period for radionuclide transport. The DF for the radioactive gaseous effluent species as it moves through KPL is a function of both the half-life and the delay bed hold-up time for the radionuclide.





The noble gases with half-lives < 1h (krypton-89m, xenon-135, xenon-135m, xenon-137 and xenon-138) will decay to insignificant levels in the delay beds due to the considerably long hold-up times in the delay beds of 40 hours for krypton and 960 hours/40 days for xenon and radioiodine. The delay beds are assumed to have no impact on carbon-14 and argon-41 discharges.

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 [KPL] relatively unaffected by any of the processing and treatment steps described. The potential for retention of carbon-14, via exchange with carbon-12 in the charcoal beds, and subsequent release during maintenance or temperature/flow fluctuations is noted [44], [47] and will be kept under review; however it is anticipated that the design and operation of the RR SMR will not result in carbon-14 accumulation in the delay beds, and this has not been considered in modelling discharges.

Total discharge from the KPL is also dependent on input volumes and activity concentrations of the coolant which is degassed. It is assumed for shutdown that the full volume of coolant {REDACTED} is completely degassed in the [KNF] degasser, and the resultant gases subsequently treated and discharged via the [KPL]. For power operations, it is conservatively assumed that a single KNF10 tank volume of coolant {REDACTED} will be degassed and routed via the [KPL] [40].

Coolant is treated by the CVCS purification system [KBE] before being transferred to [KNF] for treatment and storage. PSST activity concentrations are available for the CVCS output for the power operations and shutdown phases of operation (section 29.1.3). The CVCS PSST (shutdown) activity concentration is conservatively applied for the shutdown, outage and start up phases. The CVCS PSST (power operations), representing activity concentrations at steady state, is used to model the KPL discharges effluent during power operations. The impact of attributing the conservative shutdown activity concentration to the start-up phase will be assessed in subsequent issues of this chapter.

Discharges of gaseous radioactive effluent from the Gaseous Radioactive Effluent Treatment System [KPL] to the environment during power operation and shutdown modes over the full 18-month fuel cycle have been estimated using the MAB Reference [40]. Estimates of annual discharge are presented in Table 29.2-1, and summarised in Table 29.2-4, to allow comparison with discharges from other sources of gaseous radioactive effluent.

**Table 29.2-1: Annual Discharge from Gaseous Radioactive Effluent Treatment System [KPL]**

Radionuclide	Shutdown discharges (Bq/y)	Power operation discharges (Bq/y)	Total discharges (Bq/y)
H-3	0.00E+00	0.00E+00	0.00E+00
C-14	5.19E+08	2.24E+08	7.44E+08
Ar-41	3.65E+10	1.58E+10	5.23E+10
Kr-85	1.06E+09	2.39E+08	1.30E+09



Radionuclide	Shutdown discharges (Bq/y)	Power operation discharges (Bq/y)	Total discharges (Bq/y)
Kr-85m	3.20E+07	6.02E+06	3.80E+07
Kr-87	9.11E+00	1.71E+00	1.08E+01
Kr-88	1.82E+06	3.28E+05	2.15E+06
I-131	1.78E+07	3.35E+05	1.82E+07
Xe-131m	1.09E+08	2.50E+07	1.34E+08
Xe-133	1.64E+09	3.75E+08	2.02E+09
Xe-133m	4.76E+04	8.71E+03	5.63E+04

### 29.2.5 Discharges from the Nuclear HVAC [KL-] System

The nuclear HVAC extracts gaseous radioactive effluent arising from leakages, pool evaporation and air activation from the atmospheres of controlled areas and auxiliary areas of Reactor Island. The fraction of total radioactivity in these gaseous streams is determined by the Partition Coefficient (PC), (i.e. the ratio of the concentration of a radionuclide in the gas phase to its concentration in the liquid phase at equilibrium). The PCs for halogens and other radionuclides are derived primarily from estimates of the partitioning of radionuclides from the liquid to the steam phase in the SG tube [43] , and are therefore likely to be conservative for deriving estimates of gaseous releases to HVAC (as not all systems will operate at such high temperatures and pressures). The PCs applied in estimation of discharges to the nuclear HVAC at RD7/DRP1 are presented in Table 29.2-2. These values will be reviewed as the design and operation of the RR SMR SSCs matures.

**FA 29.2** Consideration of chemical and physical properties and behaviour of volatile species such as iodine, in operational and shutdown phases and fuel failure scenarios

**Table 29.2-2: Radionuclide Partition Coefficients (PC)**

Radionuclide group	PC	Source
Halogens	0.01	Nuclear Regulatory Report (NUREG)-17 Steam Generator (SG) partition coefficient (Table 4-41 in Reference [43] supported by findings in Reference [48].
Carbon-14	0.9	Estimate from EDF Ltd. (2020). [44], based on a range of OPEX PWR partition rates. Reference [44] indicates carbon-14 migrating to the gas phase has been shown to be lower over a full fuel cycle, however this conservative value will be applied to RR SMR as a precautionary measure.
Tritium	0.1	Based on estimates of 90 % available tritium released via liquid pathways from NUREG 0017 [43] and Reference [49]. This is considered bounding relative to ratios of predicted tritium concentration in gaseous and relative streams (e.g. ~3 % in UK EPR decision statement [50]).
Noble gases	1	Conservative assumption of 100 % noble gases in gas phase (NUREG-17 assumption noble gases rapidly transported out of the water) [43].
Other radionuclides	0.005	Conservative assumption based on the partitioning of radionuclides from the liquid to the steam phase in the SG tube (NUREG-17) [43].

The nuclear HVAC comprises a number of subsystems serving different areas of RI. The key sources of gaseous radioactive waste to nuclear HVAC are 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. Other radioactively contaminated areas (RCAs) within Reactor Island, including the containment building and auxiliary building (which contains various waste treatment and waste storage systems) also make important contribution to the nuclear HVAC [KL-] discharges.

Gaseous effluents are discharged via the main extract stack exhaust system [KLS], which is currently under development in terms of final location and design. Further details of the design and operation of the HVAC system are found in Reference [51].

The nuclear HVAC [KL-] will be designed to maintain ambient atmospheric conditions and a negative pressure across the areas it serves to prevent leakages from the buildings 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 [52]. 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 2 air changes per hour.

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. Discharges from

other potential gaseous effluent discharge routes, for example from HVAC installed in waste storage buildings will be kept under review as the design progresses but are not considered in this revision of the chapter as these facilities are at a low stage of maturity.

Consistent with RGP for gaseous discharges from nuclear power plants [52] 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. A typical removal efficiency and decontamination factor achievable for the HEPA filters in nuclear systems ( [53]) of 99.97 % (decontamination factor of 3333) is applied to the RR SMR HVAC abatement. It is noted that HEPA filters can be designed or configured to achieve DFs greater than this value, (for example [54]) and thus the application of the above value is likely to result in more conservative estimates of discharge than will be achieved by the final HVAC abatement.

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 Spent Fuel Pool and Refuelling Pool area HVAC Sub-system***

The Spent Fuel Pool [FAB10] provides storage for new, partially spent, fully spent and damaged fuel. The Spent Fuel Pool has been sized to accommodate storage of irradiated fuel for up to 10 years, to allow sufficient cooling of the fuel before it is transferred to dry casks for long term storage. Footprint within the pool is also allocated for Fuel Assembly Testing [FBA], Fuel Repair [FBB] and Fuel Cleaning [FBC] Systems [55]. 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 [56].

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, as well as provision for the storage of Reactor Pressure Vessel (RPV) upper and lower internals, and RPV in-core instrumentation during refuelling.

The Spent Fuel Pool [FAB10] is connected to the Refuelling Pool [FAF] via the Fuel Transfer Channel [FCK] and Upender Pit [FAB40.] During refuelling, the gate between the Refuelling Pool [FAF] and the Refuelling Cavity [FAE] is removed and the fuel transfer channel is opened to connect the Refuelling Pool to the Spent Fuel Pool to allow the movement of the RPV upper and lower internals, and the RPV in-core monitoring assemblies from the RPV to the Refuelling Pool, and the movement of fuel to the Upender Pit.

During power operations the Refuelling Cavity [FAE] is drained and is isolated from the Refuelling Pool [FAF] through the Refuelling Pool [FAF] gate. Only the Refuelling Pool [FAF] and the Spent Fuel Pool [FAB10] are flooded.

The Fuel Pool Purification System (FPPS) [FAL] [57] [58] is located outside containment adjacent to the Spent Fuel Pool [FAB10]. The function of the FPPS [FAL] is to maintain the chemistry specification of the coolant, including water clarity, pH control (through chemical dosing of potassium hydroxide) and removal of contaminants, including radioactive species, by backwashable filters and ion

exchange resins. Fuel Pool water levels are maintained by the Fuel Pool Supply System (FPSS) [FAT] [59], replacing evaporative losses and storage of water during refuelling or maintenance operations.

Changes in activity concentration are expected as a result of shutdown transients and mixing of pools during refuelling outage. It is therefore assumed, for the purposes of this chapter, that the mixing of these pools during shutdown operations ensures that radionuclide composition and activity concentration across the Spent Fuel Pool [FAB10], RFP [FAF] and connecting pools are in equilibrium.

The PSST for the spent fuel pool and associated pools (Fuel Pools PSST) [20] represents the activity concentration values in the fuel pool system at the start of a refuelling outage. It incorporates spikes in activity concentration from shutdown transients but before any substantial decay has occurred, and accounts for mixing of the pools at outage, including activity from the Reactor Coolant System (RCS). The Fuel Pools PSST is therefore considered to be a bounding value for calculating evaporation in both power operations and shutdown/outage discharges.

Evaporation from the Spent Fuel Pool [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 losses was performed to support the design of the Fuel Pool Supply System (FPSS) [FAT] [59].

The evaporation calculations were based on the pool design and operation at primary concept definition (PCD) but are considered bounding on the changes to pool design and operating conditions at RD7/DRP1. The final design and layout of the fuel pools are still underway, but evaporation rates will be reassessed once details of pool layout are finalised.

To estimate the activity concentration of radionuclides entrained in water vapour from the Spent Fuel Pool [FAB10] and the associated pools, the Spent Fuel Pool PSST source term activity concentration was multiplied by the relevant PC and the evaporative losses (adjusted for density) for the power operations and shutdown phases respectively. The downstream abatement provided by the HVAC HEPA filters was applied the values were then adjusted for time spent in each phase (estimated at 530 days power operations and 18 days shutdown [25]) and a correction factor applied to provide an annual discharge estimate from the total cycle length of 18 months. Full details of the calculations are provided in Reference [60].

Predicted discharges of gaseous radioactive effluent from the Spent Fuel Pool [FAB10], Refuelling Pool [FAF] and other outage pools [FAB40 and FAE] via the HVAC [KL-] system are presented in Table 29.2-4.

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

Components that contain or transport the coolant to or from the reactor core make up the Reactor Coolant System (RCS). During plant life, leakage of reactor coolant through the joint and valve interfaces of the RCS can release coolant into the atmosphere of the surrounding buildings, due to normal operational wear or mechanical deterioration. The RR SMR RCS will be designed to minimise coolant leakages during normal operations through optimised materials selection and compliance with relevant pressure system design codes [61]. For the purpose of the calculations below, operational leakage rates have been determined from OPEX and RGP for limiting system leakages to levels that do not compromise safety until RR SMR-specific leakage rates can be determined.

This methodology is based on the approach recommended by the United States Nuclear Regulatory Commission (US NRC) for assessing the release of radioactive gaseous and liquid effluents from prospective nuclear power plants. The derivation of parameters for the calculations is described in the accompanying report, NUREG-0017 [43]. The following section provides an overview of key parameters and parameter values adopted for estimating discharges originating from the identified buildings, including scaling factors or other modifications applied to the 'standard PWR' parameters to account for differences with the RR SMR design. The impact of any differences in distribution of key systems contributing to HVAC discharge between the RR SMR design and the 'standard PWR' will be considered once the RR SMR layout is confirmed.

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 relevant RR SMR PST or PSST data. The source terms for the buildings discharge calculations are selected from the available PST or PSST data most appropriate to the system in question, and reflect the phase of operation under consideration, where this information is available. The selection of PST/PSST for each building data is justified in each section.

Partitioning of radionuclides between liquid and gaseous phases is accounted for when considering activity concentrations in leakages, using the PC values from Table 29.2-2. It is considered that the radionuclide inventory in coolant leaked from the RCS during power operations will partition to the gas phase in accordance with these values on exiting the RCS and is available to be evacuated by the nuclear HVAC. For non-pressurised and low pressure systems, it is assumed that coolant leakages will be in liquid phase, which then collects in sumps and is diverted to the drainage system, with the exception of non-condensable gases (comprising noble gases and carbon-14) which are conservatively assumed to partition to the gas phase to the same degree as for the RCS, and be discharged via the HVAC.

The coolant leakage OPEX data in NUREG-0017 [43] are derived principally from observations and engineering judgement based on older PWRs. It is anticipated that leakages from the RR SMR will be comparatively lower than those presented in the NUREG-0017 report, as a consequence of application of BAT [7] to material selection, operational chemistry and operation practices. In addition, it is recognized that leakage of gases from a depressurized RCS during outage phases, or from systems operating at substantially lower pressure and temperature than the RCS will be significantly less than leakage under pressure.

In performing the analyses, the leakage rates in [43] have been corrected to account for differences in design parameters between the reference plant and the RR SMR, using adjustment factors determined from the relative thermal power, coolant mass and containment volumes of the reference plant and the RR SMR. Leakage rate adjustment factors, and the leakage rates thus derived, are presented in [60].

This approach is considered to represent a suitably conservative case for the RR SMR at the current level of design maturity. The data in [43] is the most representative leakage rate information available at this stage of design and will be used pending availability of system-specific leakage rate values for the RR SMR.

## **Containment Building**

This section describes discharges from the containment building to the nuclear HVAC and considers radioactive gaseous effluents arising from leakage of coolant from the RCS and Ar-41 gaseous effluents from the neutron activation of stable Ar-40 in the atmosphere surrounding the RPV.

Gaseous radioactive effluent arising in the containment building is derived from two sources; RCS leakage and argon-41 release from Nuclear HVAC Systems (activation and RCS leakage). In line with OPEX [43], leakage rates for non-condensable gases and other radionuclides are considered separately. A full description of the methodologies is presented in Reference [60].

### *RCS leakage of non-condensable gases*

Leakage rates for the containment building are derived for non-condensable gases and for other radionuclides, using the methodology in US NRC NUREG-0017 [43], adjusting for RR SMR design factors. The calculations are detailed in full in Reference [60].

It is assumed the gas leakage rates will apply when the RCS is at pressure. When the system is depressurised in shutdown and refuelling outage, the coolant is mixed with the spent fuel pools and associated pools for refuelling, and it is expected that leakage rates of non-condensable gases and other radionuclides will be significantly lower. The leakage rates are thus applied to power operations, and for the periods during shutdown, refuelling outage and start up phases when the RCS is above atmospheric pressure. The RCS is fully depressurised in Modes 5b, 6a and 6b, which the RR SMR outage schedule [25] indicates last for approximately 7 days out of the 18 day shutdown, refuelling outage and start-up period.

The activity of radionuclides released from coolant leakage in the containment building during power and shutdown operations is calculated using the leakage rates derived as described above and the coolant mass and activity concentration. Partition coefficients, the HVAC decontamination factor, adjustment factors for time spent in operations and shutdown phase, and a correction from cycle to annual discharge rates are applied, as described for the fuel pool evaporation discharges.

The source term used in these calculations is the PST, as the leakages are directly from the RCS. It is noted that the PST activity concentration at shutdown is applied to shutdown, start-up and outage phases, and application of the leakage rates during conditions of reduced pressure as the RR SMR prepares for shutdown refuelling outage/reactor start-up is a conservatism. Further revision of the outage schedule to align with design at RD7/DRP1 will potentially enable the estimates to be refined further and the impact of attributing the conservative shutdown activity concentration to the start up phase will be assessed in Step 3.

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

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

- 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 in the same manner as the noble gas coolant leakages described above and is exhausted to the atmosphere (along with other radionuclides in the leaked effluent) by the HVAC [KL-] system. This component is calculated as described above for other non-condensable gases.

- 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. The OPEX for argon-41 release rate is based on an average of reported releases of argon-41 from several PWR plants [43]. The predicted discharge of argon-41 from the RR SMR containment building by this mechanism has been calculated by applying the adjustment factor for thermal power presented in Reference [60] 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 from RCS leakage.

The OPEX data estimates have been retained for RD7/DRP1 calculations as there is insufficient information available on RR SMR-specific calculations for formation of argon-41 by neutron activation of argon-40 in air. The above argon-41 activation calculations will be reviewed if suitable RR SMR-specific air-activation calculations which can reasonably be applied to the whole containment volume, are available.

**FA 29.3** Consider whether RR SMR specific air activation rate modelling, and containment building volume information can be used to improve argon-41 discharge estimates.

### **Auxiliary Building**

The 'auxiliary building' category encompasses non-containment RI buildings with systems processing and storing coolant, including fluid trains, waste processing systems, fuel handling systems and storage tanks [62]

As described above, discharge to HVAC is only considered for leakages from pressurised systems or for non-condensable gases. The waste treatment systems in the auxiliary building will operate at significantly lower pressures and temperatures relative to the RCS at power (e.g. Reference [63] and [46]), to ensure abatement systems operate efficiently. It is therefore assumed that only non-condensable gases will be discharged to HVAC from the auxiliary buildings systems for all operational phases, as leakages of other radionuclides will not be in gaseous form.

Coolant is treated by the CVCS prior to letdown to the waste processing and storage systems. Therefore, the CVCS PSST activities for calculation point 9 (cold leg inlet) for power operations and shutdown phases are used in the calculations, as these represent the primary coolant activity concentration after treatment by CVCS. The CVCS PSST includes values for power operations and shutdown. To ensure a bounding estimate, the power operations PSST is applied to start-up and operational phases (c 530 days), and the shutdown PSST to the shutdown, start-up and refuelling outage phases (c. 18 days).

The auxiliary building coolant leakage rate was derived by applying the coolant mass adjustment factor and the number of days in the SMR power operating phase (ca. 530d) to the auxiliary building OPEX leakage rate.

Annual discharge of gaseous radioactive effluent from the auxiliary building via the HVAC are calculated from the leakage rate, CVCS PSST, partition coefficient, HVAC decontamination factor and adjustment and correction factors to determine annual discharges. The full calculations are provided in Reference [60].



### ***Turbine Building***

It is anticipated there will be a stand-alone HVAC system for Turbine Island, which will be developed as part of the overall design programme in 2024. For the purposes of this assessment, the information regarding emissions from this building is included here alongside the equivalent information for the RI nuclear HVAC.

Leakages of small quantities of radioactively contaminated gases to the TI may occur from SSCs which interface with RI, for example the off-gases from the turbine condenser, vent gases from the turbine gland seal, liquid and vent gases from the steam generator blowdown, and liquid and gaseous leaks into the turbine building. Decisions on the routing of TI effluent discharges and application of abatement are still to be made [64].

The source term used for these calculations is the PST. This is adopted as a conservative assumption, pending further information becoming available on the interfaces between RI and TI and consequently potential sources of leakage. Further revisions will be undertaken when a Secondary Systems Source Term (SSST) is available for TI.

The turbine building coolant leakage rate was derived by applying the coolant mass adjustment factors and the number of days in the RR SMR power operating phase (ca. 530 days) to the OPEX data for leakages from containment to the turbine building.

The leakages from the RCS to the turbine building are assumed to occur when the RCS is pressurised, as described for the containment building. It is conservatively assumed that the leakage rate applies during the power operations phase and during shutdown/outage refuelling/start-up activities where the RCS pressure is above atmospheric pressure (7/18 days, as described for the containment building). Gaseous leakages when the RCS is fully depressurised are assumed not to contribute significantly to total discharges.

The discharge contribution from coolant leakage from the turbine building was calculated from the leakage rate, PST, partition coefficient, HVAC decontamination factor and adjustment and correction factors to determine annual discharges. The full calculations are provided in Reference [60].

Predicted discharge of gaseous radioactive effluent from the identified buildings (containment, auxiliary and turbine buildings) via the HVAC [KL-] system is presented in Table 29.2-4.

## **29.2.6 Discharges from Air Removal and Evacuation System [MAJ]**

Small steam generator tube leakages (SGTLs) may result in the transfer of radioactivity from the RR SMR primary to secondary circuit, if activity levels are sufficiently low that normal operations can continue. Operational leakage limits for steam generator tube leaks and philosophy for management of SGTL's for the RR SMR have yet to be confirmed but will take into account RGP. 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 Air Removal and Evacuation System (ARES) [MAJ].

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



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 [MAJ]. The inventory of particulate and solubilised radionuclides (including tritium) in the leaked effluent are condensed and retained in the condenser aqueous phase. A fraction of the transferred radioiodine inventory partitions to the secondary circuit gas phase according to the partition coefficients listed in Table 29.2-2.

The power operations PST is the most appropriate source term for leakages from the primary coolant via a SGTL, as leakage will be direct from the RCS.

[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. A full cycle (548 days) operation is therefore assumed as a conservative basis, recognising leakages are likely to be reduced when the RCS is depressurised. This assumption is likely to be refined in future using PST from startup and outage phases and revised outage schedule information, and/or SSST data.

The primary to secondary leakage rate for [MAJ], is presented in Reference [60]. For the purposes of this analysis, it is assumed that steam generators will be standard-sized components; thus, unlike other calculations, no scaling factor has been applied to the leakage rate.

As noted, the detailed design of [MAJ] is in progress and will be developed further with vendor engagement. The above assumptions have been selected to provide a cautious estimate of discharges from [MAJ] until further design details are available.

Discharges of noble gases, carbon-14 and radioisotopes of iodine to the atmosphere from [MAJ] have been calculated using the PST activity concentration, coolant mass, leakage rate and PC across the entire operating cycle. It is assumed discharges from [MAJ] are released to the environment without abatement. A correction factor is applied for converting to annual discharges, and the estimated annual discharges from [MAJ] are presented in Table 29.2-4.

### 29.2.7 Annual Discharge of Liquid and Gaseous Radioactive Effluents

A summary of the predicted annual RR SMR liquid discharges to the environment, for all radionuclides where discharge exceeds 1 Bq/y, is presented in Table 29.2-3.

**Table 29.2-3: 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)
Ba-140	5.74E+03	I-131	2.34E+02	Pr-143	1.97E+02
C-14	8.11E+02	I-133	5.58E+01	Ru-103	3.47E+03
Ce-141	6.83E+01	K-42	9.65E+04	Sr-89	7.24E+01
Ce-143	1.93E+02	Kr-85	1.11E+01	Sr-90	6.54E+00
Ce-144	1.92E+03	Kr-85m	4.84E+06	Te-132	3.63E+02



Radionuclide	Annual discharge (Bq/y)	Radionuclide	Annual discharge (Bq/y)	Radionuclide	Annual discharge (Bq/y)
Co-58	2.41E+02	La-140	2.47E+03	Xe-131m	4.05E+06
Co-60	3.56E+01	Mn-54	9.45E+01	Xe-133	1.49E+07
Cs-134	2.07E+05	Mo-99	8.38E+00	Xe-133m	8.71E+08
Cs-136	2.45E+04	Na-24	9.18E+02	Xe-135	3.04E+05
Cs-137	1.32E+05	Nb-94	1.17E+00	Y-91	2.46E+00
Fe-55	5.21E+02	Nb-95	1.37E+02	Zn-65	1.30E+02
Fe-59	2.04E+01	Ni-59	1.35E+00	Zr-95	1.61E+02
H-3	8.58E+10	Ni-63	1.79E+02	Zr-97	4.89E+00

Table 29.2-4 presents a summary of the predicted discharges from the GRETS [KPL], HVAC [KL] and [MAJ] (note the breakdown of [KPL] discharges between operational and shutdown phases is provided in Table 29.2-1). Discharges exclude radionuclides where the discharge rate is below 1 Bq/y, in line with the liquid discharge data. Discharges from HVAC and CARS also exclude radionuclides with half-lives less than 3 minutes (1/10 of the maximum HVAC air exchange rate of 1-2 volumes per hour), which would be expected to have decayed away before release to the environment.

**Table 29.2-4: Annual Discharge from Gaseous Radioactive Effluent Treatment System [KPL], HVAC [KL-] and [MAJ]**

Radio-nuclide	Spent Fuel Pool		Containment Building		Auxiliary Building		Turbine Building		CARS	KPL	Total
	Power	Shutdown	Power	Shutdown	Power	Shutdown	Power	Shutdown	All	All	All
Ag-108m	2.62E-02	1.52E-03	1.10E-02	8.83E-02	0.00E+00	0.00E+00	1.02E-01	8.19E-01	0.00E+00	0.00E+00	1.05E+00
Ag-110m	2.01E+02	1.16E+01	8.50E+01	6.86E+02	0.00E+00	0.00E+00	7.88E+02	6.36E+03	0.00E+00	0.00E+00	8.13E+03
Ar-41	0.00E+00	0.00E+00	5.04E+11	1.51E+08	6.61E+08	8.74E+06	1.03E+10	1.37E+08	1.03E+09	5.23E+10	5.69E+11
Ba-140	9.16E-03	5.31E-04	1.04E+02	1.38E+00	0.00E+00	0.00E+00	9.65E+02	1.28E+01	0.00E+00	0.00E+00	1.08E+03
Br-84	0.00E+00	0.00E+00	1.02E+03	5.15E+01	0.00E+00	0.00E+00	9.47E+03	4.77E+02	0.00E+00	0.00E+00	1.10E+04
C-14	1.48E+08	8.58E+06	9.18E+09	1.21E+08	5.60E+08	7.40E+06	8.33E+09	1.10E+08	8.29E+08	7.44E+08	2.00E+10
Ce-141	2.96E-02	1.71E-03	3.90E+01	5.15E-01	0.00E+00	0.00E+00	3.61E+02	4.77E+00	0.00E+00	0.00E+00	4.05E+02
Ce-143	1.77E-01	1.03E-02	7.81E+02	1.03E+01	0.00E+00	0.00E+00	7.24E+03	9.57E+01	0.00E+00	0.00E+00	8.13E+03
Ce-144	3.35E-03	1.94E-04	4.22E+00	5.58E-02	0.00E+00	0.00E+00	3.91E+01	5.17E-01	0.00E+00	0.00E+00	4.39E+01
Cl-36	4.97E-04	2.88E-05	2.15E+01	2.85E-01	0.00E+00	0.00E+00	2.00E+02	2.64E+00	0.00E+00	0.00E+00	2.24E+02
Co-58	7.40E+03	4.29E+02	5.77E+02	2.56E+04	0.00E+00	0.00E+00	5.35E+03	2.37E+05	0.00E+00	0.00E+00	2.77E+05
Co-60	1.91E+02	1.11E+01	7.65E+01	6.46E+02	0.00E+00	0.00E+00	7.09E+02	5.98E+03	0.00E+00	0.00E+00	7.62E+03
Cr-51	7.12E+02	4.13E+01	6.82E+02	1.42E+03	0.00E+00	0.00E+00	6.32E+03	1.32E+04	0.00E+00	0.00E+00	2.24E+04
Cs-134	8.26E+00	4.79E-01	3.23E+02	4.14E+01	0.00E+00	0.00E+00	2.99E+03	3.83E+02	0.00E+00	0.00E+00	3.75E+03
Cs-136	8.70E+00	5.05E-01	5.31E+02	4.96E+01	0.00E+00	0.00E+00	4.92E+03	4.60E+02	0.00E+00	0.00E+00	5.97E+03
Cs-137	6.24E+00	3.61E-01	2.44E+02	3.10E+01	0.00E+00	0.00E+00	2.26E+03	2.87E+02	0.00E+00	0.00E+00	2.83E+03
Cs-138	0.00E+00	0.00E+00	3.79E+03	3.82E+02	0.00E+00	0.00E+00	3.52E+04	3.54E+03	0.00E+00	0.00E+00	4.29E+04
Fe-55	2.31E+02	1.34E+01	1.61E+02	5.58E+02	0.00E+00	0.00E+00	1.49E+03	5.17E+03	0.00E+00	0.00E+00	7.63E+03
Fe-59	9.44E+01	5.47E+00	6.84E+01	2.37E+02	0.00E+00	0.00E+00	6.34E+02	2.20E+03	0.00E+00	0.00E+00	3.24E+03
H-3	8.18E+09	4.74E+08	3.09E+09	4.09E+07	0.00E+00	0.00E+00	2.87E+10	3.79E+08	0.00E+00	0.00E+00	4.08E+10
I-131	2.05E-02	1.19E-03	3.22E+03	1.49E+02	2.04E+00	9.45E-02	2.92E+03	1.36E+02	9.68E+05	1.82E+07	1.91E+07
I-132	9.37E-09	5.43E-10	5.03E+04	4.07E+03	3.09E+01	2.49E+00	4.56E+04	3.69E+03	1.51E+07	0.00E+00	1.52E+07
I-133	3.84E-02	2.23E-03	2.72E+04	1.64E+03	1.72E+01	1.04E+00	2.47E+04	1.49E+03	8.19E+06	0.00E+00	8.25E+06
I-134	0.00E+00	0.00E+00	8.95E+04	2.43E+03	5.18E+01	1.40E+00	8.12E+04	2.20E+03	2.69E+07	0.00E+00	2.71E+07

Radio-nuclide	Spent Fuel Pool		Containment Building		Auxiliary Building		Turbine Building		CARS	KPL	Total
	Power	Shutdown	Power	Shutdown	Power	Shutdown	Power	Shutdown	All	All	All
I-135	5.88E-04	3.41E-05	5.12E+04	1.91E+03	3.22E+01	1.20E+00	4.64E+04	1.74E+03	1.54E+07	0.00E+00	1.55E+07
K-42	1.88E+01	1.09E+00	2.04E+05	2.70E+03	0.00E+00	0.00E+00	1.89E+06	2.50E+04	0.00E+00	0.00E+00	2.12E+06
Kr-85	0.00E+00	0.00E+00	4.55E+12	1.81E+11	2.89E+11	1.15E+10	4.13E+12	1.64E+11	4.11E+11	1.30E+09	9.74E+12
Kr-85m	0.00E+00	0.00E+00	3.04E+09	1.50E+08	1.86E+08	9.14E+06	2.76E+09	1.36E+08	2.74E+08	3.80E+07	6.59E+09
Kr-87	0.00E+00	0.00E+00	5.90E+09	2.38E+08	3.28E+08	1.33E+07	5.35E+09	2.16E+08	5.32E+08	1.08E+01	1.26E+10
Kr-88	0.00E+00	0.00E+00	7.58E+09	9.96E+08	4.53E+08	5.96E+07	6.87E+09	9.03E+08	6.84E+08	2.15E+06	1.75E+10
La-140	1.97E-01	1.14E-02	6.90E+02	9.13E+00	0.00E+00	0.00E+00	6.40E+03	8.46E+01	0.00E+00	0.00E+00	7.18E+03
Mn-54	1.89E+02	1.10E+01	6.44E+01	6.43E+02	0.00E+00	0.00E+00	5.97E+02	5.96E+03	0.00E+00	0.00E+00	7.46E+03
Mn-56	4.39E-05	2.54E-06	3.67E+02	1.50E+03	0.00E+00	0.00E+00	3.40E+03	1.39E+04	0.00E+00	0.00E+00	1.91E+04
Mo-99	1.02E-01	5.93E-03	5.61E+02	2.52E+01	0.00E+00	0.00E+00	5.20E+03	2.33E+02	0.00E+00	0.00E+00	6.02E+03
Na-24	6.80E-01	3.94E-02	4.14E+03	5.47E+01	0.00E+00	0.00E+00	3.83E+04	5.07E+02	0.00E+00	0.00E+00	4.30E+04
Nb-95	7.45E+01	4.32E+00	1.23E+02	1.47E+02	0.00E+00	0.00E+00	1.14E+03	1.37E+03	0.00E+00	0.00E+00	2.85E+03
Nb-97	6.74E-14	3.91E-15	1.13E+02	1.35E+02	0.00E+00	0.00E+00	1.05E+03	1.25E+03	0.00E+00	0.00E+00	2.55E+03
Ni-59	7.48E-02	4.33E-03	1.45E-01	2.52E-01	0.00E+00	0.00E+00	1.34E+00	2.34E+00	0.00E+00	0.00E+00	4.15E+00
Ni-63	1.03E+01	5.98E-01	1.99E+01	3.50E+01	0.00E+00	0.00E+00	1.85E+02	3.24E+02	0.00E+00	0.00E+00	5.75E+02
Np-237	1.48E-11	8.57E-13	1.85E-08	2.44E-10	0.00E+00	0.00E+00	1.71E-07	2.27E-09	0.00E+00	0.00E+00	1.92E-07
Np-239	1.10E-01	6.40E-03	2.87E+02	3.79E+00	0.00E+00	0.00E+00	2.66E+03	3.51E+01	0.00E+00	0.00E+00	2.98E+03
Pr-143	6.62E-02	3.84E-03	9.39E+01	1.24E+00	0.00E+00	0.00E+00	8.70E+02	1.15E+01	0.00E+00	0.00E+00	9.76E+02
Rb-88	0.00E+00	0.00E+00	1.74E+03	1.95E+02	0.00E+00	0.00E+00	1.61E+04	1.81E+03	0.00E+00	0.00E+00	1.99E+04
Rb-89	0.00E+00	0.00E+00	1.07E+03	1.20E+02	0.00E+00	0.00E+00	9.91E+03	1.11E+03	0.00E+00	0.00E+00	1.22E+04
Ru-103	1.28E-02	7.44E-04	1.68E+01	2.21E-01	0.00E+00	0.00E+00	1.55E+02	2.05E+00	0.00E+00	0.00E+00	1.74E+02
Ru-105	3.13E-05	1.82E-06	4.56E+02	6.03E+00	0.00E+00	0.00E+00	4.23E+03	5.59E+01	0.00E+00	0.00E+00	4.75E+03
Ru-106	1.89E-04	1.10E-05	2.36E-01	3.12E-03	0.00E+00	0.00E+00	2.19E+00	2.89E-02	0.00E+00	0.00E+00	2.46E+00
Sb-122	1.25E+02	7.26E+00	2.30E+01	8.06E+02	0.00E+00	0.00E+00	2.13E+02	7.47E+03	0.00E+00	0.00E+00	8.64E+03
Sb-124	5.23E+02	3.03E+01	7.00E+01	1.82E+03	0.00E+00	0.00E+00	6.49E+02	1.69E+04	0.00E+00	0.00E+00	2.00E+04

Radio-nuclide	Spent Fuel Pool		Containment Building		Auxiliary Building		Turbine Building		CARS	KPL	Total
	Power	Shutdown	Power	Shutdown	Power	Shutdown	Power	Shutdown	All	All	All
Sb-125	2.45E+01	1.42E+00	2.73E+00	8.30E+01	0.00E+00	0.00E+00	2.53E+01	7.69E+02	0.00E+00	0.00E+00	9.06E+02
Sr-89	1.98E-03	1.15E-04	2.04E+01	2.70E-01	0.00E+00	0.00E+00	1.89E+02	2.50E+00	0.00E+00	0.00E+00	2.12E+02
Sr-90	1.21E-05	7.03E-07	1.20E-01	1.59E-03	0.00E+00	0.00E+00	1.11E+00	1.47E-02	0.00E+00	0.00E+00	1.25E+00
Sr-91	2.53E-03	1.46E-04	1.87E+03	2.47E+01	0.00E+00	0.00E+00	1.73E+04	2.29E+02	0.00E+00	0.00E+00	1.94E+04
Sr-92	7.30E-08	4.23E-09	3.35E+03	4.43E+01	0.00E+00	0.00E+00	3.11E+04	4.10E+02	0.00E+00	0.00E+00	3.49E+04
Tc-99m	5.99E-03	3.47E-04	1.54E+02	7.95E+02	0.00E+00	0.00E+00	1.43E+03	7.37E+03	0.00E+00	0.00E+00	9.75E+03
Te-131	0.00E+00	0.00E+00	1.92E+03	2.54E+01	0.00E+00	0.00E+00	1.78E+04	2.35E+02	0.00E+00	0.00E+00	1.99E+04
Te-132	1.57E-02	9.08E-04	2.66E+02	3.52E+00	0.00E+00	0.00E+00	2.47E+03	3.26E+01	0.00E+00	0.00E+00	2.77E+03
Te-133m	0.00E+00	0.00E+00	2.81E+03	3.71E+01	0.00E+00	0.00E+00	2.60E+04	3.44E+02	0.00E+00	0.00E+00	2.92E+04
Te-134	0.00E+00	0.00E+00	5.05E+03	6.67E+01	0.00E+00	0.00E+00	4.68E+04	6.18E+02	0.00E+00	0.00E+00	5.25E+04
Xe-131m	0.00E+00	0.00E+00	1.86E+08	1.01E+07	1.17E+07	6.43E+05	1.68E+08	9.20E+06	1.68E+07	1.34E+08	5.37E+08
Xe-133	0.00E+00	0.00E+00	3.99E+10	1.22E+09	2.62E+09	8.29E+07	3.62E+10	1.10E+09	3.60E+09	2.02E+09	8.67E+10
Xe-133m	0.00E+00	0.00E+00	6.30E+09	3.11E+08	3.98E+08	1.96E+07	5.71E+09	2.82E+08	5.68E+08	5.63E+04	1.36E+10
Xe-135	0.00E+00	0.00E+00	3.34E+10	8.42E+08	2.84E+09	8.09E+07	3.03E+10	7.63E+08	3.02E+09	0.00E+00	7.13E+10
Xe-135m	0.00E+00	0.00E+00	1.76E+10	2.47E+09	5.82E+08	8.16E+07	1.60E+10	2.24E+09	1.59E+09	0.00E+00	4.05E+10
Xe-138	0.00E+00	0.00E+00	3.03E+10	8.58E+08	9.43E+08	2.69E+07	2.74E+10	7.78E+08	2.73E+09	0.00E+00	6.30E+10
Y-91	1.68E-02	9.77E-04	2.16E+01	2.86E-01	0.00E+00	0.00E+00	2.00E+02	2.65E+00	0.00E+00	0.00E+00	2.25E+02
Y-91m	0.00E+00	0.00E+00	2.40E+03	3.18E+01	0.00E+00	0.00E+00	2.23E+04	2.94E+02	0.00E+00	0.00E+00	2.50E+04
Y-92	1.97E-05	1.14E-06	3.11E+03	4.11E+01	0.00E+00	0.00E+00	2.88E+04	3.81E+02	0.00E+00	0.00E+00	3.23E+04
Zn-65	7.26E+00	4.21E-01	1.21E+02	4.91E+01	0.00E+00	0.00E+00	1.12E+03	4.55E+02	0.00E+00	0.00E+00	1.76E+03
Zr-95	9.83E+01	5.70E+00	1.26E+02	1.90E+02	0.00E+00	0.00E+00	1.17E+03	1.76E+03	0.00E+00	0.00E+00	3.35E+03
Zr-97	9.01E+02	5.23E+01	1.35E+04	2.03E+04	0.00E+00	0.00E+00	1.25E+05	1.88E+05	0.00E+00	0.00E+00	3.48E+05

As noted throughout this chapter, the estimates of liquid and gaseous radioactive effluent discharges are based on design information which is still subject to change, and generic assumptions based on OPEX. The quantification of discharges will be revised to align with RR SMR-specific design information or revised OPEX in place of assumptions, as the design develops.

**FA 29.4** The quantification of radioactive gaseous and liquid effluent discharges to the environment (determined by MAB and 'buildings' calculations) will be reviewed to account for design changes, including the following:

- Up-issue of the PST (including PSST/SSST)
- KNF and KPL system design
- Spent Fuel Pool and associated pools design and evaporation calculations
- HVAC design
- Buildings layout and Turbine Island design
- Air removal and evacuation system [MAJ] design and operation
- Up-issue of outage schedule
- Additional radioactive gaseous effluent discharge routes emerging from design (e.g. waste facilities)

## 29.3 Estimated Monthly Discharges

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### 29.3.1 Determination of Significant Radionuclides

Estimates of liquid and gaseous radioactive discharges to the environment for significant radionuclides are provided to support future permitting activities for the RR SMR and to enable comparison of environmental performance with other nuclear power plant (reference [17]). The significant radionuclides identified in this section provide the basis for proposed permit limits (section 29.4.3) and comparison with other nuclear power plant (section 29.5.2) within this chapter.

The EA has published a guidance document [26] setting out their criteria for setting limits on the discharge of radioactive effluents from nuclear sites. This guidance document establishes the criteria for determination of significant radionuclides from normal operations discharges. The radionuclide selection should also be consistent with Recommendation 2004/2/Euratom [65], which provides for standardised information from nuclear power plants.

The EA guidance proposes that:

“We will normally set annual site limits for each radionuclide, or group of radionuclide(s), that, for normal operation

- a. are significant in terms of radiological impact on people (that is, the dose to the most exposed group at the proposed limit exceeds 1  $\mu\text{Sv}/\text{y}$ );
- 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 40  $\mu\text{Gy}/\text{h}$ );
- c. are significant in terms of the quantity of radioactivity discharged (that is, the discharge of a radionuclide exceeds 1 TBq/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 EA’s initial radiological assessment tool (IRAT), based on the annual discharge data presented in Table 29.2-3 and Table 29.2-4. 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  $\text{m}^3/\text{s}$  selected for the GSD, and the effective release height for gaseous effluent of {REDACTED}, based on an estimated {REDACTED} height for the RR SMR shell



[66] and the assumptions presented in Reference [67] to derive a conservative value for discharge height. The full details and results of these assessments and the assumptions used in modelling radiological impact are provided in E3S Case Tier 1 Chapter 30: Prospective Radiological Assessment [68].

The radioactive effluent RR SMR discharges result in doses which are below the threshold of significance for conditions a) and b) for all individual radionuclides for both liquid and gaseous discharges. No commonly considered groups of radionuclides (e.g. radioiodines, noble gases, particulates) exceed the threshold of significance.

Based on the current predicted discharges, krypton-85 exceeds the 1 TBq threshold, and thus meets condition c) for gaseous discharges. The predicted discharge of krypton-85 from the RR SMR has been flagged as anomalous (see also section 29.5.2) and this selection will be kept under review.

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 liquid and gaseous discharges will be assessed in due course once sufficient information on the PST and RR SMR design is available.

A literature review was conducted to identify radionuclides which may be constrained under national or international agreements and although there are general obligations to reduce radiological discharges, as described in E3S Case Version 2, Tier 1, Chapter 30: Prospective Radiological Assessment [68] there are no specific requirements on individual radionuclides or groups of radionuclides predicted to be discharged from the RR SMR. The predicted discharges of radionuclides classed as 'nuclear material' by the IAEA, and requiring safeguards accountancy measures are expected to be below detectable limits. Therefore, we do not consider that condition e) for identification of significant radionuclides is met.

Increases in the concentration of fission products such as caesium-137 and tritium (given the boron-free operating chemistry for RR SMR) in liquid effluent may be indicative of fuel failure and degradation in the efficiency of abatement systems for removing particulates and therefore meet condition f). Similarly, increases in the concentrations of noble gases (other than argon-41) such as xenon-133 in gaseous effluent are also indicative of fuel failure or a degradation in the performance of the [KPL]. Excessive tritium vapours in the gaseous discharge stream may indicate faults in Spent Fuel Pool cooling or HVAC systems.

Condition g) proposes limits for the appropriate generic categories from the RSR Pollution Inventory (eg "alpha particulate" and "beta/gamma particulate" for discharges to air) to be considered. The total discharge of alpha-emitting radionuclides is expected to be very low (< 1Bq/y for all alpha-emitting radionuclides discharges by the RR SMR and is thus discounted for limits for normal operations.

In addition to consideration of the conditions above, the annual discharge limits for Sizewell B (an operating PWR power station) [69] and Hinkley Point C (a proposed PWR power station) [70] was also carried out, along with a review of the proposed limits presented in the GDA submission for the UK HPR1000 [71]. Justification for selection of significant radionuclides was in line with the categories above, and indicative of significant activities discharged (tritium, carbon-14, noble gases), radiological impact (carbon-14) and other radionuclides or radionuclide groups indicative of fuel failure, material degradation or abatement systems performance [18], [71].

Table 29.3-1 presents a summary of radionuclides identified for consideration as significant radionuclides for RR SMR.

**Table 29.3-1: Significant Radionuclides in Aqueous and Gaseous Radioactive Effluent Discharges from the RR SMR**

Criterion		Aqueous	Gaseous
a)	Public dose at proposed limit > 1 µSv/y	--	--
b)	Dose rate to non-human species at proposed limits > 40 µGy/h	--	--
c)	Discharge exceeds 1 TBq/y	--	Kr-85-
d)	Collective dose at proposed limits > 1 man-Sv /y to any population	Not assessed	Not assessed
e)	Of international concern	-	-
f)	Indicator of plant performance	Tritium, caesium-137	Tritium, xenon-133
g)	RSR reporting categories and established/ proposed limits for PWRs in the UK	Sizewell B	Tritium, caesium-137 'other radionuclides'
		Hinkley Point C	Tritium, carbon-14, cobalt-60, caesium-137, other fission and activation products
		UK HPR 1000	Tritium, carbon-14, other gamma emitters
			Tritium, carbon-14, noble gases, iodine-131, particulate beta emitters
			Tritium, carbon-14, noble gases, iodine-131, other fission & activation products
			Tritium, carbon-14, xenon-133, xenon-135, other beta/ gamma emitters

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 [65] the following are predicted to be the significant radionuclides discharged from the RR SMR during normal operations:

**Table 29.3-2: Significant Radionuclides Determined for RR SMR Discharges**

Significant Radionuclides – Liquid Discharges	Significant Radionuclides – Gaseous Discharges	
Tritium	Tritium	Xenon-133
Caesium-137	Carbon-14	Krypton-85
Other beta/gamma emitting radionuclides, (excluding tritium)	Iodine-131	Other beta/gamma emitting particulates

The source term for the RR SMR is still in development, and further design detail, e.g. abatement systems will be available as the design develops. It is also anticipated that the headroom factors will

be reviewed. As discharge estimates are updated, the list of significant radionuclides will be reviewed.

**FA29.5** List of ‘significant radionuclides’ to be reviewed when collective dose estimates are available, and as discharge estimates are revisited.

### 29.3.2 Estimation of Monthly Discharge of Radioactive Effluents

The PST 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 [56], 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:

- 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.
- For gaseous effluent discharges, it is considered that discharges over the course of the 17.5-month power operation mode will be relatively uniform, with a spike in the discharge of some radionuclides during the 18-day 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 (comprising 18-day shutdown discharges and 12-day power operation discharge).

The estimated monthly discharge of significant gaseous radionuclides is presented in Table 29.3-3 and Table 29.3-4.

**Table 29.3-3: Maximal Monthly Discharge of Aqueous Radionuclides of Significance**

Radionuclide	Annual discharge (Bq/y)	Monthly discharge (Bq/m) <sup>[1]</sup>
H-3	8.58E+10	8.58E+10
Cs-137	1.32E+05	1.32E+05
Other beta/gamma emitters (ex. tritium) <sup>[2]</sup>	4.78E+05	4.78E+05

[1] Based on the assumption the discharge is released in one month

[2] Also excludes short-lived noble gases

**Table 29.3-4: Indicative Monthly Discharge of Gaseous Radionuclides of Significance**

Status	Tritium discharge (Bq/m)	Carbon-14 discharge (Bq/m)	Iodine-131 discharge (Bq/m)	Krypton-85 discharge (Bq/m)	Xenon-133 discharge (Bq/m)	Other beta /gamma particulates (Bq/m)
Power Ops.	3.22E+09	1.58E+09	1.51E+06	7.56E+11	6.76E+09	5.53E+06
Power Ops plus Shutdown	2.21E+09	1.12E+09	9.53E+05	6.65E+11	5.54E+09	2.89E+06

The significant radionuclides list and predicted monthly discharges will be re-estimated as the design progresses and once information on collective dose is available following completion of the detailed dose assessment described in E3S Case Version 2, Tier 1, Chapter 30: Prospective Radiological Assessment [68].

## 29.4 Proposed Discharge Limits

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### 29.4.1 Introduction

Prospective operators of new nuclear power stations 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 [72]. 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.

Annual limits are normally established for selected radionuclides determined using set criteria published in Reference [26]. Section 29.3 describes how selected radionuclides for the RR SMR have been established, applying the criteria in Reference [26] supplemented with analyses of the selected radionuclides proposed in previous GDA submissions and the discharge limits set for permitted nuclear power stations in operation or under development. The remainder of this section describes proposals for the limits on discharges of the RR SMR significant radionuclides, including an account of how the limits have been determined.

### 29.4.2 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 expected events [26]. Determination of discharge limits should also take into account the requirement of the Environment Agency's (EA) radioactive substances management developed principles (RSMDP) [13] that limits should be set such that there is minimum headroom between actual levels of discharge expected during normal operation and the discharge limit.

The PST OPEX data used to determine the discharges are considered to include the contributions from expected events typical of PWRs, and thus the predicted RR SMR annual discharges presented in Table 29.2-3 and Table 29.2-4 are therefore taken to incorporate both routine discharges and the contributions from expected events. The annual limits on discharges to the environment from the RR SMR of the significant radionuclides identified in Table 29.3-1 have been determined in Reference [73] by multiplying the annual discharge from normal operations (including expected events) by a headroom factor (HF). For a radionuclide, 'i', the annual limit is derived from:

$$\text{Annual Limit}_i = (\text{Annual Discharge}_i \times \text{HF}_i)$$

Where the contribution from expected events can be disaggregated from the normal operations, it is considered that there is no need to apply a headroom factor to expected events discharges, as quantification of expected events contributions are typically based on limiting conditions/design basis activities. Derivation of the annual limit for radionuclide, 'i', then becomes:

$$\text{Annual Limit}_i = (\text{Normal Operations Discharge}_i \times \text{HF}_i) + \text{Contribution from Expected Events}_i$$

As the PST is developed further, the contribution from expected events will be assessed and disaggregated where possible to enable the estimation of annual limits to be refined.

**FA 29.6** Review, and where appropriate update, the approach for estimating the contribution of expected events to the normal operations source term and radioactive waste arisings, in consultation with the Chemistry team.

**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. Given the degree of uncertainties associated with predicted discharges from new facilities in the early years of operation (especially where such discharges are dominated by random, infrequent but expected events e.g., fuel pin failures), a reasonably generous headroom factor may be applied to discharge estimates for new facilities until operational data becomes available [26].

Previous approaches to determining HFs have included the use of standardised values [74], based on formula provided in EA guidance [75] and comparison of expected plant performance with permit limits and Operational Experience (OPEX) for similar plants [18]. More recently, GDA submissions for new nuclear power plant (NPP) designs have adopted a statistical approach for determining appropriate HFs [71], [19]. This approach applies a one-sided normal distribution at 99.9th percentile confidence interval to operational data from which the discharges were estimated. This approach is considered to provide a balance between ensuring proposed permit limits allow for variability in normal operations discharges without being excessively cautious. Table 29.4-1 presents the HFs applied in previous applications using the methods described above.

**Table 29.4-1: Headroom factors from previous NPP designs**

Radionuclide	AP1000 [74]		UK EPR <sup>a</sup> [18]		HPR1000 [71]		ABWR [19]	
	Gaseous HF	Liquid HF	Gaseous HF	Liquid HF	Gaseous HF	Liquid HF	Gaseous HF	Liquid HF
<b>Tritium</b>	Uses worst case plant discharge (WCPD) formula [75] for all liquid or gaseous radionuclide discharges to environment, applying EA-derived factor of 1.5 to account for discharge variability, and 1.1 to account for plant ageing to the estimated discharge = HF of 1.65 applied to discharges		6.00	1.44	4.55	2.58	3.8	3.8
<b>Carbon-14</b>			2.00	4.13	3.40	2.94	1.9	
<b>Radioiodines<sup>b</sup></b>			8.00	7.15	6.4		1.7	1.7
<b>Noble gases<sup>c</sup></b>			28.13		5.3		2.1	
<b>Argon-41</b>							2.9	
<b>Others<sup>d</sup></b>			30	16.67	2.89	3.11	4.1	4.1

<sup>a</sup> HF derived from proposed discharge limits/expected discharge

<sup>b</sup> Radioiodines values includes HPR1000 'Halogens' category

<sup>c</sup> Noble gases values include AWBR noble gases excluding Argon-41

<sup>d</sup> Others' values include 'particulates (AWBR) and Fission Products/Activation Products (UK EPR)

In contrast to the examples in Table 29.4-1 where the discharge estimates are based primarily on data from one or more operational sister plants, the RR SMR 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 [16]. This dataset is not sufficiently detailed to allow adoption of the statistical approach described above to the OPEX used in determining the RR SMR PST. Thus, an alternative approach to estimating headroom factors, using a generic data set from publicly available light water reactor discharge and performance data, is proposed. The methodology is described in full in Reference [73], and the data set used is described in Reference [76].

### **Radionuclide and Data Selection**

Headroom Factors were determined for key representative radionuclides/radionuclide groups, encompassing the range of radionuclide production and release mechanisms, the significant radionuclides identified in Table 29.3-2, and considering RGP and data availability.

The RR SMR is a new design and therefore there is no directly comparable OPEX on plant discharges with which to assess discharge variability. In the absence of directly comparable data, a dataset for assessing NPP discharge OPEX from broadly comparable designs, including PWR and Water-Water Energetic Reactors (VVER) designs was assembled from publicly available datasets of radioactive discharges [77], [78], [79] from light water reactors (LWR) and other design and operational information [80]. The screening and selection of the dataset, to ensure it was representative of normal operations across multiple refuelling cycles, is described in detail in Reference [76]. Discharge data were normalised to electricity production at each site (i.e. GBq/GW<sub>e</sub>) to allow comparison across sites.

The selected data was reviewed for suitability for the statistical approach outlined in [71] and [19]. It was established the datasets for the radionuclides of interest were normal or log-normal distribution and were suitable for the approach being taken. This analysis is presented in Reference [73].

### **Calculation of Headroom Factor**

Following RGP for determining HF by statistical analysis [71], [19] the mean and standard deviation are determined for each of the datasets contributing to the calculation of liquid or gaseous discharges. The HF is then determined from the equation for a one-sided normal distribution, applying a coverage factor, k, to determine the confidence level of interest.

$$HF_i = \frac{[\text{mean}_i + (k \times \text{stdev}_i)]}{\text{mean}_i}$$

Where HF<sub>i</sub> is the headroom factor for radionuclide group 'i', k is the coverage factor for selected confidence interval, and mean " and stdev "i" are the mean and standard deviation of the dataset for radionuclide group " respectively.

The previously published derivations of HF by statistical analysis have applied a confidence interval of 99.9 % to datasets gathered from comparator plants of the same or similar design. In this analysis, we have applied a confidence interval of 95 %, taking into consideration that the data in this analysis is derived from a much broader dataset. Further detail on the justification for the decision to apply the 95 % confidence interval is provided in Reference [73].

Headroom Factors were determined for each of the key radionuclides/radionuclide groups, using a confidence interval of 95% and corresponding coverage factor (k) of 1.645. The headroom factors derived for each radionuclide are recorded in Table 29.4-2 below.

**Table 29.4-2: RR SMR Headroom Factors**

	<b>Gaseous</b>	<b>Liquid</b>
<b>Radionuclide</b>	<b>HF 95 % confidence interval</b>	<b>HF 95 % confidence interval</b>
Tritium	2.52	6.29
Carbon-14	1.64	1.62
Radioiodine (I-131)	3.02	2.73
Noble gases (Xe-133)	3.98	N/A
Other radionuclides <sup>a</sup>	6.21	3.03

<sup>a</sup> 'Other radionuclides' category comprises total beta/gamma emitter activity, excluding tritium and carbon-14

### 29.4.3 Proposed Limits

Estimates of the RR SMR liquid and gaseous effluent discharges at annual limits, derived from the predicted discharges and the HF's described in section 29.4.2 are presented in Table 29.4-3 and Table 29.4-4.

**Table 29.4-3: Predicted Discharges of Liquid Radionuclides at Annual Limits <sup>a</sup>**

<b>Radio-nuclide</b>	<b>Predicted Annual Discharge (Bq/y)</b>	<b>Head room Factor</b>	<b>Discharge at Annual Limits (Bq/y)</b>	<b>Radio-nuclide</b>	<b>Predicted Annual Discharge (Bq/y)</b>	<b>Headroom Factor</b>	<b>Discharge at Annual Limits (Bq/y)</b>
Ba-140	5.74E+03	3.03	1.74E+04	Mo-99	8.38E+00	3.03	2.54E+01
C-14	8.11E+02	1.62	1.31E+03	Na-24	9.18E+02	3.03	2.78E+03
Ce-141	6.83E+01	3.03	2.07E+02	Nb-94	1.17E+00	3.03	3.53E+00
Ce-143	1.93E+02	3.03	5.85E+02	Nb-95	1.37E+02	3.03	4.16E+02
Ce-144	1.92E+03	3.03	5.81E+03	Ni-59	1.35E+00	3.03	4.10E+00
Co-58	2.41E+02	3.03	7.31E+02	Ni-63	1.79E+02	3.03	5.42E+02
Co-60	3.56E+01	3.03	1.08E+02	Pr-143	1.97E+02	3.03	5.97E+02
Cs-134	2.07E+05	3.03	6.27E+05	Ru-103	3.47E+03	3.03	1.05E+04
Cs-136	2.45E+04	3.03	7.43E+04	Sr-89	7.24E+01	3.03	2.19E+02
Cs-137	1.32E+05	3.03	4.00E+05	Sr-90	6.54E+00	3.03	1.98E+01
Fe-55	5.21E+02	3.03	1.58E+03	Te-132	3.63E+02	3.03	1.10E+03



Radio-nuclide	Predicted Annual Discharge (Bq/y)	Head room Factor	Discharge at Annual Limits (Bq/y)	Radio-nuclide	Predicted Annual Discharge (Bq/y)	Headroom Factor	Discharge at Annual Limits (Bq/y)
Fe-59	2.04E+01	3.03	6.19E+01	Xe-131m	4.05E+06	3.03	1.23E+07
H-3	8.58E+10	6.29	5.39E+11	Xe-133	1.49E+07	3.03	4.51E+07
I-131	2.34E+02	2.73	6.38E+02	Xe-133m	8.71E+08	3.03	2.64E+09
I-133	5.58E+01	2.73	1.52E+02	Xe-135	3.04E+05	3.03	9.22E+05
K-42	9.65E+04	3.03	2.92E+05	Y-91	2.46E+00	3.03	7.45E+00
Kr-85	1.11E+01	3.03	3.35E+01	Zn-65	1.30E+02	3.03	3.94E+02
Kr-85m	4.84E+06	3.03	1.47E+07	Zr-95	1.61E+02	3.03	4.89E+02
La-140	2.47E+03	3.03	7.47E+03	Zr-97	4.89E+00	3.03	1.48E+01
Mn-54	9.45E+01	3.03	2.86E+02				

<sup>a</sup> Note that there is no information which allows calculation of a separate Headroom Factor for noble gases in liquid discharge; the 'all other beta/gamma emitting nuclides' Headroom Factor has been applied.

**Table 29.4-4: Predicted Discharges of Gaseous Radionuclides at Annual Limits**

Radio-nuclide	Predicted Annual Discharge (Bq/y)	Head-room Factor	Discharge at Annual Limits (Bq/y)	Radio-nuclide	Predicted Annual Discharge (Bq/y)	Head-room Factor	Discharge at Annual Limits (Bq/y)
Ag-108m	1.05E+00	6.21	6.50E+00	Nb-95	2.85E+03	6.21	1.77E+04
Ag-110m	8.13E+03	6.21	5.05E+04	Nb-97	2.55E+03	6.21	1.58E+04
Ar-41	5.69E+11	3.98	2.26E+12	Ni-59	4.15E+00	6.21	2.58E+01
Ba-140	1.08E+03	6.21	6.73E+03	Ni-63	5.75E+02	6.21	3.57E+03
Br-84	3.67E+07	6.21	2.28E+08	Np-239	2.98E+03	6.21	1.85E+04
C-14	2.00E+10	1.64	3.29E+10	Pr-143	9.76E+02	6.21	6.06E+03
Ce-141	4.05E+02	6.21	2.52E+03	Rb-88	1.99E+04	6.21	1.23E+05
Ce-143	8.13E+03	6.21	5.05E+04	Rb-89	1.22E+04	6.21	7.58E+04
Ce-144	4.39E+01	6.21	2.73E+02	Ru-103	1.74E+02	6.21	1.08E+03
Cl-36	2.24E+02	6.21	1.39E+03	Ru-105	4.75E+03	6.21	2.95E+04
Co-58	2.77E+05	6.21	1.72E+06	Ru-106	2.46E+00	6.21	1.53E+01



Radio-nuclide	Predicted Annual Discharge (Bq/y)	Head-room Factor	Discharge at Annual Limits (Bq/y)	Radio-nuclide	Predicted Annual Discharge (Bq/y)	Head-room Factor	Discharge at Annual Limits (Bq/y)
Co-60	7.62E+03	6.21	4.73E+04	Sb-122	8.64E+03	6.21	5.37E+04
Cr-51	2.24E+04	6.21	1.39E+05	Sb-124	2.00E+04	6.21	1.24E+05
Cs-134	3.75E+03	6.21	2.33E+04	Sb-125	9.06E+02	6.21	5.63E+03
Cs-136	5.97E+03	6.21	3.71E+04	Sr-89	2.12E+02	6.21	1.32E+03
Cs-137	2.83E+03	6.21	1.76E+04	Sr-90	1.25E+00	6.21	7.77E+00
Cs-138	4.29E+04	6.21	2.66E+05	Sr-91	1.94E+04	6.21	1.21E+05
Fe-55	7.63E+03	6.21	4.74E+04	Sr-92	3.49E+04	6.21	2.16E+05
Fe-59	3.24E+03	6.21	2.01E+04	Tc-99m	9.75E+03	6.21	6.05E+04
H-3	4.08E+10	2.52	1.03E+11	Te-131	1.99E+04	6.21	1.24E+05
I-131	1.91E+07	3.02	5.78E+07	Te-132	2.77E+03	6.21	1.72E+04
I-132	1.52E+07	3.02	4.60E+07	Te-133m	2.92E+04	6.21	1.81E+05
I-133	8.25E+06	3.02	2.49E+07	Te-134	5.25E+04	6.21	3.26E+05
I-134	2.71E+07	3.02	8.19E+07	Xe-131m	5.37E+08	3.98	2.14E+09
I-135	1.55E+07	3.02	4.68E+07	Xe-133	8.67E+10	3.98	3.45E+11
K-42	2.12E+06	6.21	1.32E+07	Xe-133m	1.36E+10	3.98	5.41E+10
Kr-85	9.74E+12	3.98	3.88E+13	Xe-135	7.13E+10	3.98	2.84E+11
Kr-85m	6.59E+09	3.98	2.62E+10	Xe-135m	4.05E+10	3.98	1.61E+11
Kr-87	1.26E+10	3.98	5.01E+10	Xe-138	6.30E+10	3.98	2.51E+11
Kr-88	1.75E+10	3.98	6.98E+10	Y-91	2.25E+02	6.21	1.40E+03
La-140	7.18E+03	6.21	4.46E+04	Y-91m	2.50E+04	6.21	1.55E+05
Mn-54	7.46E+03	6.21	4.64E+04	Y-92	3.23E+04	6.21	2.01E+05
Mn-56	1.91E+04	6.21	1.19E+05	Zn-65	1.76E+03	6.21	1.09E+04
Mo-99	6.02E+03	6.21	3.74E+04	Zr-95	3.35E+03	6.21	2.08E+04
Na-24	4.30E+04	6.21	2.67E+05	Zr-97	3.48E+05	6.21	2.16E+06

The annual discharge limits proposed for significant radionuclides in aqueous and gaseous effluents discharged from the RR SMR are shown in Table 29.4-5.



**Table 29.4-5: Proposed Annual Limits for Radioactive Discharges from RR SMR**

Discharge phase	Radionuclide/ radionuclide group	Predicted annual discharges (Bq/y )	Proposed annual (12 month rolling average) Limit (Bq)
Aqueous	H-3	8.58E+10	5.39E+11
	Cs-137	1.32E+05	4.00E+05
	Other beta/gamma	4.54E+05	1.37E+06
Gaseous	H-3	4.08E+10	1.03E+11
	C-14	2.00E+10	3.29E+10
	I-131	1.91E+07	5.78E+07
	Xe-133	8.67E+10	3.45E+11
	Kr-85	9.74E+12	3.88E+13
	Other beta/gamma particulate	6.93E+07	2.20E+08

The predicted discharges and proposed annual limits will be revised as the PST develops, including availability of system specific PSST/SSST, the disaggregation of expected events contribution from normal operations source term, and additional details on waste system design are available from vendor engagement.

**FA 29.7** Review, and where appropriate, update, the approach to generation of headroom factors, including the dataset used, treatment of the data and confidence intervals applied to the data.

## 29.5 Comparison with Discharges from Similar Facilities

### 29.5.1 Overview

The UK Discharge Strategy [12] 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 (Table 29.2-3 and Table 29.2-4) have been normalised (to 1 GW<sub>e</sub>) and compared to normalised discharges from selected PWRs.

The analysis of liquid and gaseous radioactive effluent discharges from operational LWRs was based on annual discharge data and related information published in publicly accessible online platforms ([80], [77], [78], [79]). For completeness, the predicted discharges from previous GDA candidate reactor designs (References [74], [19] and [71]) have also been included in Table 29.5-1 and Table 29.5-2 and compared with the predicted RR SMR discharges. Full details of the methodology for identification of suitable comparator reactors, radionuclide selection, collation of discharge data, and normalisation of discharges across reactors is presented in Reference [76].

The comparisons presented below will be reassessed as the RR SMR discharge estimates are developed further.

**FA. 29.8** The comparison of discharge estimates against existing nuclear power stations will be revised as the design of the RR SMR progresses and calculation of disposals updated accordingly.

### 29.5.2 Comparison of Predicted and Reported Annual Discharges

#### Liquid Discharges

Table 29.5-1 presents the normalised discharges (GBq/GWh) of representative radionuclides in aqueous effluent reported for the shortlisted reactor sites averaged over the data reference period and the predicted annual discharges for GDA candidate reactor designs. The data for the GDA reactors is shown in pale grey cells, and that for RR SMR is shown in **bold**.

**Table 29.5-1: Average of Normalised Aqueous Discharges (GBq/GWh)**

Reactor site	Tritium	Carbon-14	Caesium-137	Iodine-131	Total Beta/ Gamma (excl. tritium & carbon-14)
Kozloduy	1.5E+00		3.8E-06	1.3E-06	1.3E-05
Temelin	2.9E+00		2.6E-06	9.3E-08	1.2E-05
Cattenom	6.0E+00	2.8E-03	2.5E-06	1.3E-06	2.9E-03
Chooz B	2.7E+00	1.3E-03	1.6E-06	4.6E-07	1.4E-03
Civaux	3.2E+00	1.6E-03	1.6E-06	2.1E-07	1.7E-03
Golfech	3.2E+00	1.6E-03	5.8E-07	4.5E-07	1.7E-03

Reactor site	Tritium	Carbon-14	Caesium-137	Iodine-131	Total Beta/ Gamma (excl. tritium & carbon-14)
Penly	3.2E+00	2.4E-03	7.2E-07	3.7E-07	2.4E-03
Mochovce	1.3E+00		3.3E-07	6.3E-08	2.1E-06
Watts Bar	1.4E+01		5.8E-08		3.4E-05
Seabrook	4.1E+00		2.1E-09		3.8E-05
Comanche Peak	3.8E+00				2.6E-06
Sizewell B	2.4E+00		5.9E-04		1.2E-03
HPR1000	2.6E+00	1.5E-03			3.2E-05
AP1000	3.8E-03	3.8E-04	2.6E-06	1.7E-06	2.7E-04
UK EPR	3.4E+00	1.5E-03	3.7E-06	4.6E-07	
UK ABWR	1.7E-02		1.4E-10	3.0E-09	2.6E-07
<b>RR SMR</b>	<b>2.2E-02</b>	<b>2.1E-10</b>	<b>3.4E-08</b>	<b>5.9E-11</b>	<b>1.2E-07</b>

The predicted discharges of aqueous radioactive effluent from the RR SMR are well below the maximum values recorded for the shortlisted reactor sites and GDA designs assessed. RR SMR registered the lowest normalised discharges of carbon-14, iodine-131 and total beta/ gamma (excl. tritium & carbon-14) on a normalised basis. Normalised discharges of tritium and caesium-137 are well below the reported discharges from the operating reactors and within the range of predicted discharge values for other GDA candidate reactors.

The observed minimal discharge of aqueous radioactive effluent is a consequence of the design approach and operational philosophy that is being applied to the RR SMR, which drives the incorporation of plant features and configurations that will maximise the reuse of fluids and effluents and minimise total discharges to the environment. This approach is consistent with/ implements the fundamental principles of concentrating and containing radioactivity, and the use of BAT to minimise radiological impacts, over the RR SMR lifecycle. One of the key RR SMR design features that enables the accomplishment of this minimal discharge objective is the elimination of soluble boron and lithium from the primary coolant during normal operations, which significantly reduces the generation of tritium – a limiting factor for reuse of primary coolant effluents. Other enablers of this approach include the adoption of abatement techniques (e.g. reverse osmosis (RO) and evaporation) that allow preferential removal and concentration of contaminants and the recycling of the ‘clean’ effluent.

### **Gaseous Discharges**

Table 29.5-2 presents the normalised discharges (GBq/GWh) of representative radionuclides in gaseous effluent reported or predicted for the shortlisted reactor sites and GDA candidate reactor designs. The data for the GDA reactors is shown in pale grey cells, and that for RR SMR is shown in **bold**.

**Table 29.5-2: Average of Normalised Gaseous Discharges (GBq/GWh)**

Reactor site	Tritium	Carbon-14	Iodine-131	Total Noble Gas	Total Iodines (incl. I-131)	Total Beta/Gamma (particulates)
Cattenom	1.8E-01	4.4E-02	1.6E-6	8.2E-02	4.0E-06	4.4E-07
Chooz B	3.8E-02	1.9E-02	1.9E-06	5.8E-02	2.9E-06	2.7E-07
Civaux	8.0E-02	1.6E-02	1.2E-06	5.2E-02	2.6E-06	8.9E-08
Golfech	5.5E-02	2.1E-02	1.1E-06	1.5E-01	4.9E-06	1.7E-07
Penly	4.4E-02	2.8E-02	7.5E-07	2.9E-02	2.4E-06	2.1E-07
Sizewell B	5.8E-02	2.8E-02	1.4E-06	3.1E-01	1.3E-06	1.8E-06
Kozloduooy	3.4E-02	3.9E-02	5.1E-07	5.8E-02	5.1E-07	2.0E-06
Mochovce	7.5E-02	6.2E-02	4.4E-06	1.3E-01	8.2E-06	1.9E-06
Temelin	1.0E-01	3.9E-02	3.5E-06	3.5E-01	3.6E-06	4.7E-06
Watts Bar	2.0E-01	4.1E-02	7.2E-09	5.4E-01	8.1E-08	1.4E-07
Seabrook	3.2E-01	4.4E-02	1.4E-08	2.5E-03	1.4E-08	8.5E-09
Comanche Peak	6.5E-02	4.6E-02	-	8.5E-04	2.5E-10	-
HPR1000	8.1E-02	3.6E-02		1.2E-01	1.7E-06	3.7E-07
AP1000	2.1E-01	6.9E-02	2.4E-05	7.6E-01	6.4E-05	1.9E-06
UK EPR	3.3E-02	2.3E-02	1.5E-06	5.3E-02	3.3E-06	2.6E-07
UK ABWR	2.3E-01	7.7E-02	1.6E-05	1.5E-01	2.7E-05	2.1E-08
<b>RR SMR</b>	<b>1.1E-02</b>	<b>5.4E-03</b>	<b>5.2E-06</b>	<b>2.9E-00</b>	<b>1.6E-05</b>	<b>8.8E-07</b>

As is evident from Table 29.5-2 predicted discharges of tritium and carbon-14 in gaseous effluent are significantly less than the reported discharge of these radionuclides in comparator reactors. Iodine-131/total radioiodines, and total beta/gamma particulates are within the range of those found at comparative sites. In contrast, the predicted discharges of noble gases is conspicuously greater than those from comparator reactors.

The greater normalised activity of noble gases observed in RR SMR discharges is attributable to high activity concentration of krypton-85 in the primary coolant, which is orders of magnitude greater than the activity of other noble gases in the coolant.

Noble gas activity concentration in the RD7/DRP1 PST [20] is derived from OPEX data, which is conservatively assumed to represent steady-state conditions within a fuel cycle. The longer half-life of krypton-85 means steady state is not reached, but continually increases over the 18-month cycle. This results in a greater range of OPEX data and an over-estimate of predicted krypton-85 activity concentration in the coolant, which is translated into comparatively high discharges to atmosphere. This is confirmed by an initial comparison of normalised discharge of noble gases excluding krypton-85, which brings the RR SMR well within the range of discharges from other comparator reactors.

The methodology for estimating krypton-85 coolant concentrations for Issue 2 of the PST will be revised so that it better represents steady state conditions.

It is noted that there is some variability in the make-up of the radionuclide categories from the comparator reactors, especially total beta/ gamma. For the RR SMR, this radionuclide category comprises >50 particulate radionuclides (the bulk of which are of the order of a few kBq or less). For the shortlisted operating reactors, the total beta/gamma discharges comprise >30 particulate radionuclides. Iodine-134 is not reported for comparator reactors and has therefore been excluded from the total iodines for RR SMR.

## 29.6 Conclusions

### 29.6.1 ALARP, BAT, Secure by Design, Safeguards by Design

The implementation of the conduct design engineering process in the design of all structures, systems and components (SSC) involving handling, generation or processing of radioactive waste, ensures that discharges of radioactive liquid and gaseous effluent from the RR SMR are kept as low as reasonably practicable (ALARP), are optimised through Best Available Techniques (BAT), and implement secure by design and safeguards by design principles. This ensures that the RR SMR E3S Principles are applied throughout. The decision-making process for each SSC, demonstrating how ALARP and BAT have been considered during the design, is captured in the associated engineering decision record.

This approach supports the chapter top-level claim ‘SMR waste arisings during normal operation are well understood and defined, with discharges optimised using BAT and minimised to ALARA’, and subclaim ‘Discharges of aqueous and gaseous radioactive effluent to the environment from the RR SMR are reduced to levels that are ALARA’. Note that ALARA and ALARP are considered to be equivalent in meaning for the purposes of this chapter.

The ongoing development of the design, analysis, and verification of SSCs, as described throughout the E3S case ensures the E3S fundamental objective can be met by the RR SMR at all life cycle phases and design stages, demonstrating that risks are or will be reduced to ALARP, that BAT is applied, and the design incorporates secure by design and safeguards by design principles.

### 29.6.2 Assumptions & Commitments on Future Dutyholder

The E3S Case will capture assumptions and commitments for future dutyholders/permit holders/licencees. The following assumptions and commitments in relation to discharges of radioactive waste have been identified in this revision of the E3S Case and are listed in Table 29.6-1. This list will be kept under review and updated in future revisions of the E3S Case.

**Table 29.6-1: Assumptions and Commitments on Future Dutyholders/Permit-holders/Licensees**

Assumption/Commitment	ID	Description
Future Dutyholder / Licensee/Permit Holder will determine proposed permit limits for liquid and gaseous effluent discharges from normal operations	C29.1	It is a requirement of the EPR16 RSR permitting process that the applicant provides proposed permit limits based on expected discharges. The limits proposed in this chapter are indicative of the RR SMR discharges in normal operations, based on current understanding of the PST and RR SMR design, but will need to be reviewed to ensure they are consistent with proposed operation of a specific plant.
Future Dutyholder/Permit Holder/Licensee will establish a means of determining liquid and gaseous effluent	C29.2	Permit holders are required to demonstrate compliance with effluent discharge limits set by the EA. This may be through monitoring, sampling (see E3S Case Tier 1 Chapter 28: Sampling and Monitoring Arrangements



Assumption/Commitment	ID	Description
discharges from normal operations are in compliance site permit limits.		[81]), or calculation as agreed with the Environment Agency.
Future Dutyholder/ Permit Holder/ Licensee will review normal site operations discharges after a period of time.	C29.3	The EA require permit limits to be set with the minimum headroom to allow for variability in normal operations. The predicted discharges presented in this chapter are anticipated to be conservative estimates based on OPEX and cautious assumptions about RR SMR performance. EA would expect to review discharge limits after a period of operation to ensure limits were still appropriate.

### 29.6.3 Conclusions and Forward Look

The generic E3S Case objective at Version 2 is ‘to provide confidence that the RR SMR design will be capable of delivering the E3S fundamental objective as it developed from a concept design into a detailed design’. This confidence is built through development and underpinning of top-level claims across each chapter of the E3S Case, through supporting arguments and evidence. The top-level claim for E3S Case Version 2, Tier 1, Chapter 29: Quantification of Radioactive Effluent Discharges and Prospective Limits is ‘SMR waste arisings during normal operation are well understood and defined, with discharges optimised using BAT and reduced and minimised to ALARA’.

The arguments and evidence presented to meet the generic E3S Case objective at Version 2 include the proposed approaches and methodologies for estimating the liquid and gaseous radioactive effluent discharges and proposed permit limits for the normal operation of the RR SMR. The discharge estimates thus derived are compared with discharges from similar LWR plant in Europe and the US.

Liquid radioactive effluent discharges were found to be below the average of reported or forecasted discharges from comparable LWRs. The normalised annual discharge of gaseous radioactive effluent from the RR SMR is broadly consistent with the average of forecast or reported discharges from other PWR plants. The predicted radioactive effluent discharges are derived using conservative assumptions made in relation to parameters such as efficiency of abatement technology or radionuclide speciation and are expected to be reduced as the reactor design develops.

Results of an initial dose assessment, using the Environment Agency’s IRAT and the conservative assumptions regarding dispersion of liquid and gaseous discharges to the environment are presented in E3S Case Version 2, Tier 1, Chapter 30: Prospective Radiological Assessment [68]. This assessment indicated that the total dose to a representative member of the public from all relevant exposure pathways is less than 0.007 mSv/y. This is well below the source dose constraint of 0.3 mSv/y [11] and the more restrictive constraint of 0.15 mSv/y recommended to be applied at the design stage of new nuclear facilities by the Health Protection Agency (now UK Health Security Agency) [82]. The dose rate to the worst affected organism was estimated at 0.00 5µGy/h, which is orders of magnitude below the statutory guideline value of 40 µGy/h [13]. Details of the initial assessment of radiological impacts due to predicted discharges of radioactive effluent from the RR SMR are presented in E3S Case Version 2, Tier 1, Chapter 30: Prospective Radiological Assessment [68].

The interim calculations and assessments will be re-evaluated as the design of the RR SMR progresses. The forward actions listed in Table 29.6-2 of this chapter summarise the key actions which will be undertaken to support the production of the subsequent version(s) of this chapter to be issued in future revisions of the E3S case.

The estimates of discharges of gaseous and liquid radioactive effluent provided in this chapter are based on RR SMR design at RD7/DRP1, supported by OPEX derived from similar nuclear power stations and industry standard treatment systems. Further work will be undertaken to refine the estimates, based on the evolving RR SMR design, along with ongoing testing and development work, and continued review of relevant literature and published discharge data.

The forward actions (FA) in Table 29.6-2 will support the E3S case and demonstrate that the RR SMR will be compliant with BAT, ALARP and secure by design principles and meet key objectives and assessment criteria. The forward actions below are a summary of the key actions required; detailed actions can be found in supporting documents including [33], [40] [60] [73] and [76].

**Table 29.6-2: Forward Action Plan for Quantification of Discharges from RR SMR**

ID	Description	Date
FA 29.1	The Fault Schedule shall be reviewed periodically to ensure all expected events are accounted for, and, where possible, their contribution is quantified	Q4 2024 and ongoing
FA 29.2	Consideration of chemical and physical properties and behaviour of volatile species such as iodine, in operational and shutdown phases and fuel failure scenarios, and the impact on discharges	Before E3S Case Version 3
FA 29.3	Consider whether RR SMR specific air activation rate modelling, and containment building volume information can be used to improve argon-41 discharge estimates	Before E3S Case Version 3
FA 29.4	<p>The quantification of radioactive gaseous and liquid effluent discharges to the environment (determined by MAB and ‘buildings’ calculations) will be reviewed to account for design changes, including the following:</p> <ul style="list-style-type: none"> <li>Up-issue of the PST (including PSST/SSST)</li> <li>KNF and KPL system design</li> <li>Spent Fuel Pool and associated pools design and evaporation calculation</li> <li>HVAC design</li> <li>Buildings layout and Turbine Island design</li> <li>Air removal and evacuation system [MAJ] design and operation</li> <li>Up-issue of outage schedule</li> <li>Additional radioactive gaseous effluent discharge routes emerging from design (e.g. waste facilities)</li> </ul>	<ul style="list-style-type: none"> <li>Q2 2025</li> <li>Q3 2024</li> <li>Q1 2025</li> <li>Q1 2025</li> <li>Q1, Q4 2024</li> <li>Q4 2024</li> <li>Ongoing</li> </ul>



FA 29.5	List of 'significant radionuclides' to be reviewed when collective dose estimates are available, and as discharge estimates are revisited	Q3 2024; see FA29.1
FA 29.6	Review, and where appropriate update, the approach for estimating the contribution of expected events to the normal operations source term and radioactive waste arisings, in consultation with the Chemistry team.	Before E3S Case Version 3
FA29.7	Review, and where appropriate, update, the approach to generation of headroom factors, including the dataset used, treatment of the data and confidence intervals applied to the data.	Before E3S Case Version 3
FA 29.8	The comparison of discharge estimates against existing nuclear power stations will be revised as the design of the RR SMR progresses and calculation of disposals updated accordingly	Before E3S Case Version 3

## 29.7 References

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- [1] Rolls-Royce SMR Limited SMR0004294 Issue 1, “Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 1: Introduction,” March 2023.
- [2] Rolls-Royce SMR Limited SMR0002131 Issue 2, “RR SMR Integrated Waste Strategy,,” 2023.
- [3] Rolls-Royce SMR Limited SMR0004502 Issue 3, “Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 11: Management of Radioactive Waste,” May 2024.
- [4] Rolls-Royce SMR Limited SMR0004139 Issue 3, “Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 12: Radiation Protection,” May 2024.
- [5] Rolls-Royce SMR Limited SMR0004982 Issue 3, “Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 20: Chemistry,” May 2024.
- [6] Rolls-Royce SMR Limited SMR0010327 Issue 1, “Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 25: Detailed Information about the Design,” May 2024.
- [7] Rolls-Royce SMR Limited SMR0008133 Issue 2, “Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 27: Implementation of BAT,” May 2024.
- [8] Rolls-Royce SMR Limited SMR0002155 Issue 3, “E3S Case CAE Route Map,” November 2023.
- [9] ICRP, ““Publication No. 103: The 2007 Recommendations of the International Commission on Radiological Protection .,” Annals of the ICRP , vol. 37, no. 2-4,” 2007.
- [10] “Council Directive 2013/59/EURATOM laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation and repealing Directives 89/618 Euratom, 90/641/Euratom, 97/43/Euratom and 2003/122/Euratom,,” 2013.
- [11] The Stationary Office, “The Environmental Permitting (England and Wales) Regulations (UK Statutory Instruments 2016 No. 1154),” 2016.
- [12] Department of Energy & Climate Change, “ Statutory Guidance to the Environment Agency concerning the regulation of radioactive discharges into the environment,” 2009.
- [13] Environment Agency, “Guidance: Radioactive substances regulation (RSR): objective and principles,,” December 2021. [Online]. Available: Online]. Available: <https://www.gov.uk/government/publications/radioactive-substances-regulation-rsr-objective-and-principles>.
- [14] Environment Agency, “Regulatory Guidance Series RSR 1: Radioactive Substances Regulation – Environmental Principles (Version 2),” 2010. [Online].
- [15] Rolls-Royce SMR, “GDA Scope and Submission Plan for Generic Environment Report SMR0004252, draft,” 2023.
- [16] Rolls-Royce SMR Limited SMR0000867 Issue 1, “Normal Operation Source Term Strategy Report, Issue 1,” 2022.
- [17] Environment Agency, “Guidance, New nuclear power plants: Generic Design Assessment guidance for Requesting Parties,” [Online]. Available: <https://www.gov.uk/government/publications/new-nuclear-power-plants-generic-design-assessment-guidance-for-requesting-parties/new-nuclear-power-plants-generic-design-assessment-guidance-for-requesting-parties#information-required-for-environment-case-subm>. [Accessed July 2022].
- [18] AREVA NP SAS/EDF, “PCER Sub-Chapter 6.3 - Outputs for the operating installation UKEPR-0003-063 Issue 5,” 2012.

- [19] Hitachi GE, “UK ABWR Generic Design Assessment, Quantification of Discharges and Limits (GA91-9901-0025-00001 Rev.G),” 2017.
- [20] Rolls-Royce SMR Limited, “Phase 2 Primary System Source Term,” December 2023.
- [21] Rolls-Royce SMR Limited SMR0000861 Issue 1, “Normal Operation Source Term Radionuclide Selection Report,” 2022.
- [22] Rolls-Royce SMR Limited SMR0001603 Issue 1, “Environment, Safety Security and Safeguards Design Principles,” August 2022.
- [23] Rolls-Royce SMR Limited SMR0000512 Issue 2, “Small Modular Reactor Radioactive Source Term Policy,” 2022.
- [24] Rolls-Royce SMR Limited SMR0004247 Issue 3, “Environment, Safety, Security and Safeguards Case Tier 1 Chapter 13: Conduct of Operations,” May 2024.
- [25] Rolls-Royce SMR Limited SMR0007699 Issue 1, “RR SMR Outage Schedule,” 2023.
- [26] Environment Agency, “Criteria for setting limits on the discharge of radioactive waste from nuclear sites,” 2012.
- [27] WENRA, “Safety Objectives for New Power Reactors. Study by WENRA Reactor Harmonization Working Group,” 2009.
- [28] IAEA, “Safety Analysis and Licensing Documentation for Nuclear Fuel Cycle Facilities (Safety Report Series No. 102),” 2020.
- [29] US NRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition – Transient and Accident Analysis (NUREG-0800, Chapter 15),” [Online]. Available: <https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/ch15/index.html>. [Accessed 26 November 2023].
- [30] CNSC, “REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants,” [Online]. Available: <https://www.cnsccsn.gc.ca/eng/acts-and-regulations/regulatory-documents/published/html/regdoc2-5-2/index.cfm>. [Accessed 26 November 2023].
- [31] ONR, “NS-TAST-GD-006 Issue 5.1 Design Basis Analysis,” 2022.
- [32] Atkins Limited EDNS010000957314 Issue 1, “Definition of expected non-normal events to be included in the environment,” 2021.
- [33] Rolls-Royce SMR Limited SMR0009430 Issue 1, “Review of Postulated RR SMR Expected Events,” November 2024.
- [34] Rolls-Royce SMR Limited SMR0001389 Issue 3, “Rolls Royce SMR Definition of Postulated Initiating Event and Derivation of Initiating Event Frequencies,” 2023.
- [35] Rolls-Royce SMR Limited SMR0004916 Issue 1, “Rolls-Royce SMR Fault Schedule (Version 6),” 2023.
- [36] IAEA, “NF-T-2.5 Review of Fuel Failures in Water cooled Reactors (2006-2015): An Update of IAEA Nuclear Energy Series no NF-T-2.1,” 2019.
- [37] Rolls-Royce SMR Limited SMR0007902, “RQ01074 Spent Fuel Arisings - Full Response,” 2023.
- [38] Rolls-Royce SMR Limited SMR0000631 Issue 2, “System Outline Description for the Processing and Treatment System for Radioactive Liquid Effluent (KNF),” June 2023.
- [39] Rolls-Royce SMR Limited SMR0000633 Issue 2, “Waste Systems Tanks Sizing Note,” October 2022.
- [40] Rolls-Royce SMR Limited SMR0006974 Issue 1, “Quantification of Radiological Discharges During Normal Operations - Reactor Waste Systems,” November 2023.
- [41] Rolls-Royce SMR Limited EDNS01000910874 Issue 1, “[KNF] Vacuum Degasser Sizing Note,” February 2021.

- [42] Rolls-Royce SMR Limited SMR0003323 Issue 1, "Waste Treatment Systems Mass and Activity Balance (MAB) Tool," May 2023.
- [43] US NRC, "Calculation of Release of Radioactive Materials in Radioactive Effluents from Pressurised Water Reactors (NUREG-0017, Revision 2)," 2020.
- [44] EDF Energy Ltd, "Variation Technical Supplement, Environmental Permit EPR/XB3539DH (SZB/THR/169, Rev. 000)," 2020.
- [45] Rolls-Royce SMR Limited SMR000632 Issue 2, "SMR000632\_2 System Design Description Reactor Island Drainage System [KTA]," November 2023.
- [46] Rolls-Royce SMR Limited SMR0000746 Issue 2, "System Design Definition for the Gaseous Radioactive Effluent Treatment System [KPL]," June 2023.
- [47] Environment Agency, "Permitting decision XB3539DH-V007," 2021.
- [48] A. Tigeras, M. Bachet, H. Hatalette and S. E., "WR iodine speciation and behaviour under normal primary coolant conditions: An analysis of thermodynamic calculations, sensibility evaluations and NPP feedback," *Progress in Nuclear Energy*, vol. 53, no. 5, pp. 504-515, 2011.
- [49] O. Sepall and S. Mason, "Vapour/Liquid partition of tritium in tritiated water," *Canadian Journal of Radioactivity*, pp. 2040 -50, 1960.
- [50] Environment Agency, "Generic design assessment: UK EPR™ nuclear power plant design by Electricité de France SA and AREVA NP SAS. Final Assessment Report - Gaseous radioactive waste disposal and limits," 2011.
- [51] Rolls-Royce SMR Limited SMR0004702 Issue 1, "System Design Description for the HVAC systems serving controlled areas and uncontrolled areas of Reactor Island [KL]," June 2023.
- [52] Office for Nuclear Regulation, "Ventillation, ONR Technical Assessment Guide (TAG) NS-TAST-GD-022," 2022.
- [53] IAEA, "Particulate Filtration in Nuclear Facilities IAEA TRS 325," 1991.
- [54] W. Bergman, M. W. First, W. L. Anderson, H. Gilbert and J. W. Jacox, "riteria for Calculating the Efficiency of HEPA Filters During and After Design Basis Accidents (23rd DOE/NRC Nuclear Air Cleaning and Treatment Conference)," 1994.
- [55] Rolls-Royce SMR Limited SMR0000983 Issue 2, "System Outline Description for the Handling of Nuclear Equipment [F] system," March 2023.
- [56] Rolls-Royce SMR Limited SMR0000594 Issue 1, " RR SMR Design Overview Report," 2022.
- [57] Rolls-Royce SMR Limited SMR0000574 Issue 1, "System Outline Description for the Fuel Pool Purification System [FAL]," April 2022.
- [58] Rolls-Royce SMR Limited SMR0005677 Issue 1, "System Design Description for the Fuel Pool Fluid Systems," April 2023.
- [59] Rolls-Royce SMR Limited SMR0000578 Issue 1, "System Outline Description for the Fuel Pool Supply System [FAT]," May 2022.
- [60] Rolls-Royce SMR Limited SMR0009429 Issue 2, "Quantification of Radiological Discharges During Normal Operations - HVAC and CARS," May 2024.
- [61] Rolls-Royce SMR Limited SMR0004363 Issue 3, "Environment, Safety, Security and Safeguards Case Tier 1 Chapter 23: Structural Integrity," May 2024.
- [62] Rolls-Royce SMR Limited SMR0009029 Issue 1, " RD7 Design Report," November 2023.
- [63] Rolls-Royce SMR Limited SMR0000631 Issue 2, "System Design Description for Liquid Radioactive Effluent Processing system [KNF]," June 2023.
- [64] Rolls-Royce SMR Limited SMR0003417 Issue 1, "T01-037 Turbine Island Decision Summary - Vents and Drains," March 2023.

- [65] European Commission, "COMMISSION RECOMMENDATION of 18 December 2003 on standardised information on radioactive airborne and liquid discharges into the environment from nuclear power reactors and reprocessing plants in normal operation," January 2004. [Online]. Available: <https://eur-lex.europa.eu/LexUriServ/LexUriServ.do?uri=OJ:L:2004:002:0036:0046:EN:PDF>. [Accessed September 2022].
- [66] Atkins Ltd , "EDNS01000961241 D46 Architectural Reports," 2021.
- [67] Rolls-Royce SMR Limited SMR0010055 Issue 1, "Inputs and Assumptions for RR SMR Radiological Dose Modelling - Continuous Discharge Scenario," February 2024.
- [68] Rolls-Royce SMR Limited SMR0004490 Issue 3, "Environment, Safety, Security and Safeguards Case Tier 1 Chapter 30: Prospective Radiological Assessment," May 2024.
- [69] Food Standards Agency, "Radioactivity in Food and the Environment (RIFE)," [Online]. Available: <https://www.food.gov.uk/research/radioactivity-in-food-and-the-environment>. [Accessed 14 July 2023].
- [70] Environment Agency, "Application by NNB Generation Company Ltd (NNB GenCo) to carry on radioactive substance activities at Hinkley Point C Power Station EPR/ZP3690SY/A001 – Decision Document,," 2013.
- [71] General Nuclear Systems Ltd, "UK HPR1000 GDA, Pre-Construction Environmental Report Chapter 6 Quantification of Discharges and Limits [HPR/GDA/PCER/0006," 2021.
- [72] Environment Agency, "Part RSR-B3 Guidance Notes (Version 4)," 2021.
- [73] Rolls-Royce SMR Limited SMR0009431 Issue 1, " Estimation of Headroom Factors for RR SMR Proposed Limits," February 2024.
- [74] Westinghouse Electric Company, "U.K. AP1000 Plant Environment Report," 2017.
- [75] Environment Agency, "SC010034/SR Developing guidance for setting limits on radioactive discharges to the environment from nuclear licensed sites," 2005.
- [76] Rolls-Royce SMR Limited SMR0009432 Issue 2, "Comparison of Discharges from RR SMR and Other Light Water Reactors," May 2024.
- [77] European Commission, "Radioactive Discharges Database (RADD)," September 2023. [Online]. Available: <https://europa.eu/radd/index.dox>.
- [78] United States Nuclear Regulatory Committee, "US NRC Radioactive Effluent and Environmental Reports," [Online]. Available: <https://www.nrc.gov/docs/ML2213/ML22133A009.pdf>. [Accessed September 2023].
- [79] Environment Agency, Food Standards Agency, Natural Resources Wales and Scottish Environment Protection Agency, "Radioactivity in food and the environment (RIFE) reports," [Online]. Available: Radioactivity in food and the environment (<https://webarchive.nationalarchives.gov.uk/ukgwa/20231023170654/https://www.gov.uk/government/publications/radioactivity-in-food-and-the-environment-rife-reports..>) [Accessed September 2023].
- [80] IAEA, "Power Reactor Information System (PRIS)," [Online]. Available: <https://pris.iaea.org/pris/About.aspx>. [Accessed September 2023].
- [81] Rolls-Royce SMR Limited: SMR0010323 Issue 2, "Environment, Safety, Security and Safeguards Case Tier 1 Chapter 28: Sampling and Monitoring Arrangements," May 2024.
- [82] Health Protection Agency, "Application of the 2007 Recommendations of the ICRP to the UK: Advice from the Health Protection Agency. Documents of the Health Protection Agency, Radiation, Chemicals and Environmental Hazards," 2009.

## 29.8 Appendix A

### 29.8.1 Claims, Arguments and Evidence

Table 29.8-1 provides a mapping of how the EA information requirements [17] link to high-level chapter claims, and how these links to the corresponding sections of the chapter that summarise the arguments and/or evidence. The full decomposition of the claims and the link to underpinning Tier 2 and Tier 3 information containing the detailed arguments and evidence is presented in the E3S case route map [8].

**Table 29.8-1: Mapping of Claims to Evidence**

EA Requirement	High-level Claims	Information Source
Quantitative estimates of waste arisings for normal operation are required	Quantitative estimates of discharges and disposals of aqueous and gaseous effluents are defined, bounded by the RR SMR normal operations source term.	<u>Chapter 29</u> , Sections 29.3 and 29.4 (gaseous and aqueous radioactive wastes) <u>Chapter 11</u> (management of radioactive wastes)
For gaseous and aqueous radioactive waste, estimate the monthly discharges:	Discharges of aqueous and gaseous radioactive effluent to the environment from the RR SMR are reduced to levels that are ALARA.	<u>Chapter 29</u> , Section 5
Estimates of discharges and disposals should clearly show the contribution of: <ul style="list-style-type: none"> <li>• Routine operations</li> <li>• Start-up and shutdown</li> <li>• Maintenance and testing</li> <li>• Infrequent but necessary aspects of operation</li> <li>• Foreseeable (based on a fault analysis) unplanned events during normal operation that remain consistent with using BAT</li> </ul>	Radioactive effluent discharges from the RR SMR, corrected for differences in design, are comparable to discharges from similar nuclear power stations globally.	<u>Chapter 29</u> , Sections 29.3 & 29.4
Estimates supported with performance data from similar facilities, where such facilities exist. Explain, where relevant, how changes in design or operation from those facilities affect the		<u>Chapter 29</u> , Section 29.5





EA Requirement	High-level Claims	Information Source
expected discharges and disposals.		
Demonstrate that discharges and waste arisings will not exceed those of comparable power stations across the world [12].		<u>Chapter 29, Section 29.7</u> (gaseous and aqueous radioactive effluents) <u>Chapter 26</u> (solid radioactive waste for offsite disposal)
Proposed discharge limits (gaseous, aqueous, disposal of combustible waste by onsite incineration)		<u>Chapter 29, Section 29.6</u> (gaseous and aqueous discharges) <u>Chapter 26</u> (combustible waste)
Proposals for annual site limits (on a rolling 12-month basis) for gaseous and aqueous discharges.		<u>Chapter 29, Section 29.6</u>

## 29.9 Glossary of Terms and Abbreviations

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ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrence
AP	Activation Products
AP1000	Advanced Passive 1000 Reactor
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
ActP	Actinide Products
BAT	Best Available Techniques
BE	Best Estimate
BSS	Basic Safety Standards
BWR	Boiling Water Reactor
CAE	Claims Arguments Evidence
CDS	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
DF	Decontamination Factor
DOORS	Dynamic Object Oriented Requirements System
DRP	Design Reference Point
E3S	Environment, Safety, Security and Safeguards
EA	Environment Agency
EC	European Commission
EE	Expected Event
EPR	European Pressurised Reactor
EPR16	Environmental Permitting (England and Wales) Regulations 2016
FA	Forward Action

[FAB10]	Spent Fuel Pool
[FAB40]	Uppender Pit
[FAE]	Refuelling Cavity
[FAF]	Refuelling Pool
[FBA]	Fuel Assembly Testing Station
[FBB]	Fuel Assembly Repair Station
[FBC]	Ultrasonic cleaning station
[FCK]	Fuel Transfer Channel
FP	Fission products
FPPS [FAL]	Fuel Pool Purification System
FPSS [FAT]	Fuel Pool Supply System
GDA	Generic Design Assessment
GDPs	Generic Developed Principles
GBq	Giga Becquerel
GRETS [KPL]	Gaseous Radioactive Effluent Treatment System
GWe	Giga Watts electrical
HF	Headroom Factor
HEPA	High-Efficiency Particulate Air (filter)
HPR1000	Hualong Pressurised Reactor
HVAC [KL-}	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiation Protection
IEF	Initiating Event Frequency
IRAT	Initial Radiological Assessment Tool
IX	Ion Exchange
LWR	Light Water Reactors
LMDS [KPL]	Liquid Effluent Monitoring and Discharge System
MAB	Mass and Activity Balance
mSv	Milli Sievert



NPP	Nuclear Power Plant
NRW	Natural Resources Wales
NUREG	Nuclear Regulatory Report
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
RCDT	Reactor Coolant Drain Tank.
RCS	Reactor Coolant System
RGP	Relevant Good Practice
RI [R01]	Reactor Island
RO	Reverse Osmosis
ROPs	RSR Objective and Principles
RPV	Reactor Pressure Vessel
RR SMR	Rolls-Royce Small Modular Reactor
RSMDP	Radioactive Substances Management Developed Principles
RSR	Radioactive Substances Regulations
SG	Steam Generator
SGBD	Steam Generator Blowdown
SMR	Small Modular Reactor
SSCs	Structures, Systems and Components
SSST	Secondary System Source Term
TI {T01}	Turbine Island
UK	United Kingdom
US	United States



US NRC	US Nuclear Regulatory Commission
VRF	Volume Reduction Factors
VVER	Water-Water Energetic Reactor
WCPD	Worst Case Plant Discharge
WENRA	Association of Regulators of Western Europe
WWTP[GM-]	Wastewater Treatment Plant
$\mu\text{Sv}$	Micro Sieverts
$\mu\text{Gy}$	Micro Grays