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UK ABWR

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UK ABWR Generic Design Assessment

Quantification of Discharges and Limits

Hitachi-GE Nuclear Energy, Ltd.

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1. Acronyms

ABWR	Advanced Boiling Water Reactor
ActP	Actinide Product
AGR	Advanced Gas cooled Reactor
ALARA	As Low As Reasonably Achievable
AP	Activation Product
BAT	Best Available Technique
BE	Best Estimate
Bq	Becquerel
BWR	Boiling Water Reactor
CA	Cycle Average
CAD	Controlled Area Drain System
CD	Condensate Demineraliser System
CF	Condensate Filter System
CFDW	Condensate and Feedwater System
CONW	Concentrated Waste System
CP	Corrosion Product
CPS	Condensate Purification System
CST	Condensate Storage Tank
CUW	Reactor Water Clean-up System
D/S	Dryer/Separator
DB	Design Basis
DF	Decontamination Factor
EIC	Environmental Impact Control
EPR	European Pressure Reactor
EPR16	Environmental Permitting (England and Wales) Regulations 2016
EPRI	Electric Power Research Institute
EU	European Union
EUST	End User Source Term
f-value	Fuel leakage rate
FDW	Feedwater System
FP	Fission Product
FPC	Fuel Pool Cooling and Clean-up System
GB	Great Britain
GDA	Generic Design Assessment

GEP	Generic Environmental Permit
Gy	Gray
HCW	High Chemical Impurity Waste System
HD	Feedwater Heater Drain System
HEPA	High Efficiency Particulate Air System
HPCP	High Pressure Condensate Pump
HVAC	Heating Ventilation and Air Conditioning System
IAEA	International Atomic Energy Agency
ILW	Intermediate Level Waste
LCO	Limiting Control for Operation
LCW	Low Chemical Impurities Waste System
LD	Laundry Drain System
LLW	Low Level Waste
LPCP	Low Pressure Condensate Pump
NPP	Nuclear Power Plant
NRW	Natural Resources Wales
OG	Off-Gas System
ONR	Office for Nuclear Regulation (UK)
OPEX	Operating Experience Feedback
ORIGEN	Oak Ridge Isotope Generator (Code)
P&ID	Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs
PO	Power Operation
PST	Primary Source Term
PRIS	Power Reactor Information System
PrST	Process Source Term
PWR	Pressurized Water Reactor
R/B	Reactor Building
RADD	RadioActive Discharges Database
RCA	Radiologically Controlled Area
RCCV	Reinforced Concrete Containment Vessel
RD	Radioactive Drain Transfer System
REP	Radioactive Substances Regulation – Environmental Principle
RGP	Relevant Good Practice
RHR	Residual Heat Removal System
RI	Regulatory Issue

RO	Regulatory Observation
RPV	Reactor Pressure Vessel
RSR	Radioactive Substances Regulation
Rw/B	Radwaste Building
S/B	Service Building
S/C	Suppression Chamber
S/P	Suppression Pool
SFP	Spent Fuel Storage Pool
SJAE	Steam Jet Air Ejector
SS	Spent Sludge System
ST	Source Term
Sv	Sievert
T/B	Turbine Building
TGS	Turbine Gland Steam System
UK	United Kingdom
US	United States

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3. Introduction

The UK requirements for the quantification of discharges and the proposing of limits as part of the GDA process are defined within the Process and Information Document (P&ID) [Ref-1]. This report provides a summary of the information and data on the gaseous and liquid discharges anticipated from the normal operation of the UK ABWR, and is in line with the information requirements outlined in the P&ID.

Initial information on discharges was provided to the Environment Agency by Hitachi-GE (Hitachi-GE Nuclear Energy Ltd.) at Step 1b of the GDA process to enable meaningful discussions to take place during Step 2. As part of these interactions, additional work was undertaken within Hitachi-GE to further define and justify the radioactive content of the UK ABWR (known as 'source term') used for the quantification of radioactive discharges within the UK ABWR. A Regulatory Observation (RO) [Ref-2] on source term was also jointly raised by the Office for Nuclear Regulation (ONR) and the Environment Agency during Step 2. This was followed by a Regulatory Issue (RI) during Step 3 [Ref-3]. In responding to the Source Term RO and RI Hitachi-GE undertook a substantial amount of further work during Step 3.

Additionally an RO relating to Turbine Gland Steam (TGS) system [Ref-4] was raised during the Environment Agency's assessment of the Step 3 submission. In response, Hitachi-GE undertook an assessment of discharges from the TGS system and also confirmed whether there are additional minor discharge routes that need to be added in the source term [Ref-5]. This document has been updated to reflect the responses to the TGS RO and the latest source term updates resulting from the Source Term.

4. Regulatory Context

4.1. P&ID Requirements

The P&ID requirements relating to the quantification of discharges and limits are reproduced from Table 1 of [Ref-1] below:

‘Quantification of radioactive waste disposals.

Provide quantitative estimates for normal operation of:

• discharges of gaseous and aqueous radioactive wastes; .....

..... 'Normal operation' includes the operational fluctuations, trends and events that are expected to occur over the lifetime of the facility, such as start-up, shutdown, maintenance, etc. It does not include increased discharges arising from other events, inconsistent with the use of BAT, such as accidents, inadequate maintenance, and inadequate operation.

For gaseous and aqueous radioactive wastes, estimate your monthly discharges:

• on an individual radionuclide basis for significant radionuclides;

• on a group basis (for example 'total alpha' or 'total beta') for other radionuclides;

• via each discharge point and discharge route.

'Significant ' radionuclides are those which:

• are significant in terms of radiological impact for people or non-human species;

• are significant in terms of the quantity of radioactivity discharged (that is, numerically high);

• have long half-lives, may persist and/or accumulate in the environment, and may contribute significantly to collective dose;

• are significant indicators of facility performance and process control.

The radionuclide selection should be consistent with reference EU, 2004…

…Estimates of discharges and disposals should clearly show the contribution of each constituent aspect of normal operations, including:

• routine operation (that is, typically, the design basis or "flowsheet design" and the minimum level of disposals);

• start-up and shutdown;

• maintenance and testing;

• infrequent but necessary aspects of operation, for example, plant wash-out; and the foreseeable, undesired deviations from planned operation (based on a fault analysis) consistent with the use of BAT, for example, occasional fuel pin failures.

Support your estimates with performance data from similar facilities and explain, where relevant, how changes in design or operation from those facilities affect the expected discharges and disposals. You should demonstrate that discharges and waste arisings will not exceed those of comparable power stations across the world (as required by UK Government policy (GB

Parliament, 2008)).

Provide your proposed limits for:

• gaseous discharges;

• aqueous discharges;

• disposal of combustible waste by on-site incineration.

Provide your proposals for annual site limits (on a rolling twelve months basis) for gaseous and aqueous discharges, and monthly limits for disposals by on-site incineration, and describe how these were derived. If desired, additionally propose limits to reflect an operating cycle, that is, 'campaign' limits.'

4.2. Alignment with the REPs

The methodologies presented in this report are consistent with industry Relevant Good Practice (RGP) and take into account the relevant Radioactive Substances Regulation – Environmental Principles (REPs) [Ref-17]. Hitachi-GE's 'Alignment with the Radioactive Substances Regulation Environmental Principles' report [Ref-18] details the approach undertaken by Hitachi-GE to reviewing and taking account of the relevant REPs within the GDA submission.

The REPs considered most relevant to the quantification of discharges and limits, as far as is covered in the scope of this GDA report, are Fundamental Principle E (Protecting Human Health and the environment) and Generic REP RSMDP12.

4.3. Regulatory Framework

The Environmental Permitting (England and Wales) Regulations 2016 (EPR16) [Ref-13] applies limits and constraints on the annual radiation exposure of members of the public. The principal aims of the legislation are that the Environment Agency, in exercising its duties and functions under the regulations, ensures 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 as low as reasonably achievable (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 1mSv per year;
- The individual dose received from any new discharge source since 13th May 2000 does not exceed 0.3mSv per year; and
- The individual dose received from any single site does not exceed 0.5mSv per year.

The 2009 'Statutory Guidance to the Environment Agency concerning the regulation of radioactive discharges to the environment' [Ref-14] provides a lower bound of exposure for the most exposed group of members of the public of 10μSv per year, below which the Environment Agency should not seek to further reduce the discharge limits that are in place, provided that the holder of the permit continues to apply Best Available Techniques (BAT).

The generation, treatment, management and disposal of all radioactive waste that will arise as result of the operation of the UK ABWR have been assessed in order to demonstrate that BAT has been applied, with the results presented in the "Demonstration of BAT report" [Ref-15] and the "Approach to Sampling and Monitoring report" [Ref-16].

5. Source Terms

5.1. General Description

The concentration of each radionuclide in the reactor water and reactor steam within the primary system (i.e. the reactor core and coolant) is known as the 'Primary Source Term' (PST). The radioactive concentrations in the ancillary systems are known as the 'Process Source Term' (PrST). The PST and PrST are used to derive and assess the radiological doses and radionuclide activity associated with all aspects of plant operation and lifecycle, including radiation protection, radioactive waste, decommissioning, and discharge assessments (these are termed 'technical areas' and each has its own associated 'End User Source Term' or EUST) [Ref-9]. In this document the EUST of interest is that for "discharge assessment".

The PST and PrST are based on modelling and operating experience feedback (OPEX) data from modern generating plants which have comparable design and operational philosophy; and thus are pertinent to the UK ABWR Source Term assessment. For the UK ABWR, relevant data has been gathered from a number of operating modern BWRs as well as the ABWRs operating in Japan.

The PST methodology report [Ref-6] describes the generation of radioactivity in reactor water and reactor steam for Best Estimate (BE) and Design Basis (DB) conditions to give corresponding BE and DB source term values, where a BE value represents a realistic level of radioactivity likely to arise during normal operation and a DB value represents a more conservative level:

i. The BE source term value provides a realistic assessment of the activity relating to a radionuclide and corresponds to a realistic level of radioactivity likely to be observed during the different phases of normal operation of the UK ABWR (Power Operation, Start-up, Shutdown, Outage etc). The BE value will be used to inform design decisions such as environmental discharges and radioactive waste disposability assessments in the technical area of radioactive waste (radwaste).
ii. The DB source term value provides a conservative assessment of the activity relating to a radionuclide and corresponds to an upper level of radioactivity likely to be observed during the different phases of normal operation of the UK ABWR, as well as during expected events (where "expected events" are anticipated but unplanned events likely to happen during the lifetime of the reactor). As with the BE value, the DB value will be used to inform design decisions in the UK ABWR, that have safety-related functions such as shielding design in the technical area of radiation protection.

Both the DB and BE types of source term can represent the activity in the system at any given point, at any given phase of the operating cycle. By contrast, Cycle Average (CA) source term values represent average levels of activity observed at any given point, but over a full reactor cycle i.e. CA source term values take into account the radioactivity levels contributed by all operating modes (i.e. Start-up, Power Operation, Shutdown and Outage) at once.

iii. CA values can be calculated for both BE (realistic) and DB (conservative) conditions. The CA is used to calculate the radioactivity in parts of the system where activity is 'held' for periods of time (e.g. in tanks) and so the immediate effect of different peaks and troughs in levels of radioactivity are buffered. CA source term values take account of influences associated with the different operating modes, for example the fission product spike in the PST at shutdowns.

The methodology report [Ref-6] also defines the Limiting Condition for Operation (LCO) for the source term of a fuel pin failure in addition to the BE, DB, CA values.

The LCO and DB values of Fission Product take into account the increase in activity brought about due to a fuel pinhole leak and thus, a fuel pin failure. The key parameter of interest where a fuel pin failure occurs is the *f*-value, where the *f*-value is defined as the release rate of noble gases, and thus the release rate associated with xenon and krypton. An *f*-value corresponding to the LCO is defined for the UK ABWR as a practical LCO value based on Japanese operating experience feedback (OPEX). The LCO for the UK ABWR is set at $1x10^8$ Bq/s; the DB for the UK ABWR is set at $1x10^9$ Bq/s.

Where relevant, the different EUSTs take account of expected events; in the case of the EUST for discharge assessment this covers the additional activity released during the event of fuel pin failure (which is the only expected event considered to have an impact on discharges - see section 7.1.6).

In summary, the 'End User Source Term' for discharge assessments is based upon the 'Best Estimate' Source Term, and also accounts for a discrete fuel pin failure expected event (using the 'LCO' value) for calculating optimised discharge amounts.

There is no intention to dispose of combustible waste by on-site incineration. Therefore discharge via this route has not been considered.

5.2. Production Mechanisms

The radionuclides present in the reactor pressure vessel of the UK ABWR, i.e. in reactor water and steam within the reactor pressure vessel, fall into three distinct categories based on their production mechanism within the reactor core:

- Fission products (FP) and Actinides products (ActP);
- Corrosion products (CP);
- Activation products (AP).

The principal means of generation of these radionuclides in the reactor is described in the subsections below.

5.2.1. Fission Products and Actinides Products

The presence of fission products and actinides products in the reactor water and reactor steam are mainly due to two mechanisms a) fissioning of trace amounts of uranium that may be present on the external surfaces of the fuel assemblies, so called "tramp uranium" and b) leakage of volatile fission products though small pinhole defects in the cladding surrounding the fuel during irradiation in the core. Typical FPs include radioisotopes of Krypton, Xenon and Iodine, Caesium-137 and Strontium-90. Typical actinides include: Americium-241 and Plutonium-239.

5.2.2. Corrosion Products

Soluble and insoluble "particulate" corrosion products are formed as materials dissolved into the reactor water, or particulate that arises as part of the normal wear and tear of reactor operations, are carried by the reactor water through the intense neutron field present in the reactor core. This results in the activation of the suspended or solubilised material to produce radioactive corrosion products such as Cobalt-60 and Iron-55/59. Particulates such as activated corrosion products are produced in the core and are retained within the water circuit, i.e. they are not carried over into the steam circuit.

5.2.3. Activation Products

The neutron activation of the reactor water will also produce radioactive species. Of particular interest are Carbon-14, Argon-41 and Tritium; the main production mechanisms for these nuclides are described in the following paragraphs.

<u>Carbon-14 (C-14)</u>
C-14 is mainly produced by O-17 (n,a) C-14 reaction in the reactor water. The C-14 produced as a result of activation of the reactor water is likely to be present as CO_2 as conditions in the upper part of the reactor core are oxidising and is transferred to the reactor steam.

<u>Argon-41 (Ar-41)</u>
Ar-41 is produced by the Ar-40 (n.g) Ar-41 reaction in reactor water. The reactor water contains a very

small amount of residual entrained air, which itself contains trace amounts of naturally occurring stable argon; the argon becomes activated as the reactor water passes through the reactor core. The Ar-41 so produced is transferred to the reactor steam.

Tritium (H-3)
H-3 is mainly produced by activation of naturally occurring deuterium (H-2) in the reactor water, H-2(n,g)H-3. H-3 is assumed to partition equally between steam and water. H-3 is unaffected by demineralisation or filtration.

Since H-3 has a relatively long half-life (12 years) and will not be affected by clean-up processes in the plant systems (notably demineralisation) the concentration will be controlled by the rate of loss of water by evaporation (especially from the Spent Fuel Storage Pool(SFP)) or by leakage / drains. All plant process water and steam will have a common H-3 concentration and the concentration reached will depend on the water /evaporation loss rate.

5.3. Radionuclides Considered

The full list of radionuclides expected to be produced in the UK ABWR's PST was generated through use of the Oak Ridge Isotope Generator (ORIGEN) Code, giving a raw output of over 600 radionuclides. To give a meaningful and useful assessment of the UK ABWR's discharges, this list was 'down selected' to about one hundred nuclides from the perspective of each EUST technical area (see section 5.1) [Ref-8].

For the information and data on the gaseous and liquid discharges anticipated from normal operation, the radionuclides considered are selected through:

- considering the practical requirements of the discharges assessment i.e. which radionuclides are important as significant contributors to dose and activity.
- incorporating a review of regulatory guidance on which radionuclides should be considered for assessment.

The review of regulatory guidance focused upon two documents:

- The European Commission document "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" 2004/2/Euratom [Ref-19], which provides a comprehensive list of radioisotopes to be assessed for liquid or airborne discharges from nuclear power plants; and,
- The Environment Agency's 'Criteria for setting limits on the discharge of radioactive waste from nuclear sites' June 2012 [Ref-22], which is used to determine the radionuclides for which limits will be set.

The radionuclides therefore considered for the UK ABWR's gaseous release are:

Ar-41	Xe-133m	Fe-59	Sb-122	Ce-144	I-131
Kr-85	Xe-135	Co-60	Sb-124	Pu-238	I-132
Kr-85m	Xe-135m	Zn-65	Sb-125	Pu-239	I-133
Kr-87	Xe-137	Sr-89	Cs-134	Pu-240	I-135
Kr-88	Xe-138	Sr-90	Cs-137	Am-241	H-3
Kr-89	Cr-51	Zr-95	Ba-140	Cm-242	C-14
Xe-131m	Mn-54	Nb-95	La-140	Cm-243	
Xe-133	Co-58	Ag-110m	Ce-141	Cm-244	

The radionuclides therefore considered for the UK ABWR's liquid releases are:

Cr-51	Ni-63	Ru-103	Sb-125	Ce-141	Cm-242
Mn-54	Zn-65	Ru-106	I-131	Ce-144	Cm-243
Fe-55	Sr-89	Ag-110m	Cs-134	Pu-238	Cm-244
Fe-59	Sr-90	Sb-122	Cs-137	Pu-239	H-3
Co-58	Zr-95	Te-123m	Ba-140	Pu-240	
Co-60	Nb-95	Sb-124	La-140	Am-241	

5.4. Operating Modes Considered

As introduced in section 5.1, the UK ABWR source term addresses all stages of the normal operating cycle, which includes several modes. This is because the radioisotopes produced are dependent on the operating mode of the reactor. The discrete reactor cycle modes associated with the UK ABWR under normal commercial operation, known as Operating Condition I, are described in [Ref-20] and summarised below. It does not include upset, emergency, fault, or testing conditions.

The operating modes of the UK ABWR during Operating Condition I are:

- **Start-up:** At the beginning of this mode, the reactor is in a shutdown mode with all control rods inserted. Once the mode switch is selected to 'start-up' the control rods are withdrawn from the core and the reactor is taken critical. When the mode switch is selected 'run', this mode moves on to Power Operation mode.
- **Power Operation:** In this mode, the reactor is critical. This mode covers power states from start-up. Based on an 18 month reactor cycle with 17 months at power, this phase is by far the longest duration at approximately 517 days.
- **Hot Stand-by:** In this mode, the reactor is sub-critical (effective neutron multiplication factor (k_{eff}) < 0.95) and coolant temperature is higher than 100°C.
- **Shutdown:** The shutdown process takes the plant from hot stand-by to the cold shutdown when reactor coolant temperature is lower or equal to 100°C with the reactor sub-critical (effective neutron multiplication factor (k_{eff}) < 0.95). In some Source Term assessments, an unplanned shutdown and subsequent re-start is assumed to occur during one full operating cycle.
- **Outage:** Once the plant is in a cold shutdown condition, re-fuelling operations may begin. This involves the flooding of the reactor well, removal of the vessel head and upper internals (steam separator and dryer) and removal of fuel to the SFP. New fuel is introduced and the plant is made ready for start-up. During refuelling outage, many essential maintenance operations are carried out. For the UK ABWR, the outage phase is anticipated to last for approximately 30 days.

The routes of discharge to environment are also affected by the operating mode of the UK ABWR. These are outlined in section 6.

6. Discharge Routes

Schematic outlines showing the general routes of gaseous and liquid discharges during the different operating modes are presented in Figure 6-1 and Figure 6-2. Hitachi-GE has identified additional minor discharge routes in [Ref-5] and these routes have been incorporated in this document.

This document considers the gaseous and liquid discharges that arise from the operation of the reactor steam and water systems. Gaseous discharges associated with the Off-Gas (OG) system, the Heating Ventilation and Air Conditioning (HVAC) system (including contributors such as the SFP), the TGS system via the Main Stack are considered. The HVAC system discharge also includes the tank vents/extracts from the various tanks in the Radwaste Building (Rw/B), which joins the Rw/B HVAC system ducting before it exits the Rw/B. Liquid discharges from the Reactor Water Clean Up system via a single point to the sea are considered. Discharges from the following minor sources are outside the scope of GDA and so not considered in this report (these discharges will be assessed at the site-specific stage where relevant):

- The dry-solid LLW processing system, the ILW store and the interim spent fuel store; these facilities are at concept design stage only and information on the precise level of discharges will not be available until the site specific stage, it is expected that the discharges from these facilities will make up only a small fraction of the overall site discharges;
- The Service Building (S/B) HVAC system; the S/B HVAC system draws from controlled areas including the laundry and active laboratory, and the discharge is made through a vent in the S/B roof. Future operators will decide on the precise plant and equipment which they wish to house in the S/B and a separate re-assessment of what is in the HVAC discharge may need to be made at site specific permitting stage. An initial assessment of the gaseous discharges from the S/B HVAC system shows that the annual activity discharged via this route is negligible [Ref-10].

The key differences between gaseous discharges during power operations, start-up and shutdown compared to those at outage are as follows:

- There is no discharge via the OG system and the TGS system during outage (as the reactor is shut down during outage).
- There are additional discharges via the HVAC system during outage. The increased discharges arise because of increased evaporation due to the temporarily elevated SFP temperature as reactor water mixes with the fuel pool water and fuel is transferred from the reactor, as well as additional discharge routes from the Reactor Well, Dryer/Separator (D/S) pool, the Suppression Pool and the Turbine Condenser Hotwell.

Liquid discharges are less sensitive to the operating mode as the discharges are collected in bulk tanks prior to periodic discharge to sea. The tank contents are sampled prior to discharge: if the content complies with the various permit conditions then a discharge is made; if the permit conditions are not met then the tank contents are recycled through clean-up systems and sampled until it can be shown the permit conditions are met.

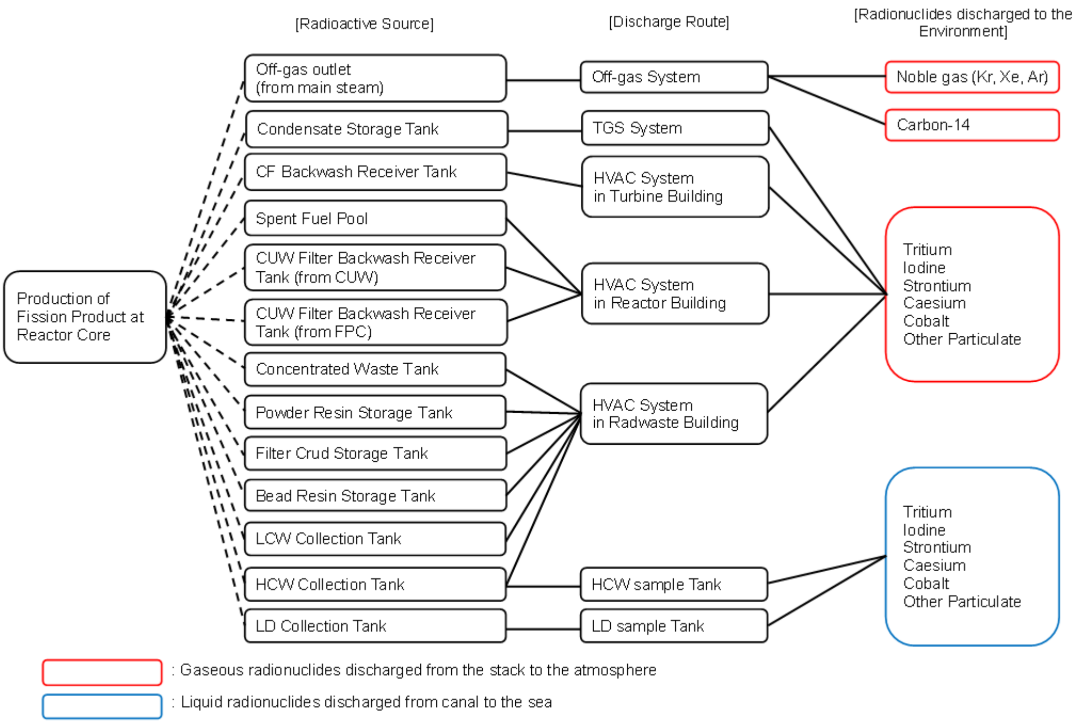

Figure 6-1: Radioactive Discharge Routes to the Environment during Power Operation, Start-up and Shutdown

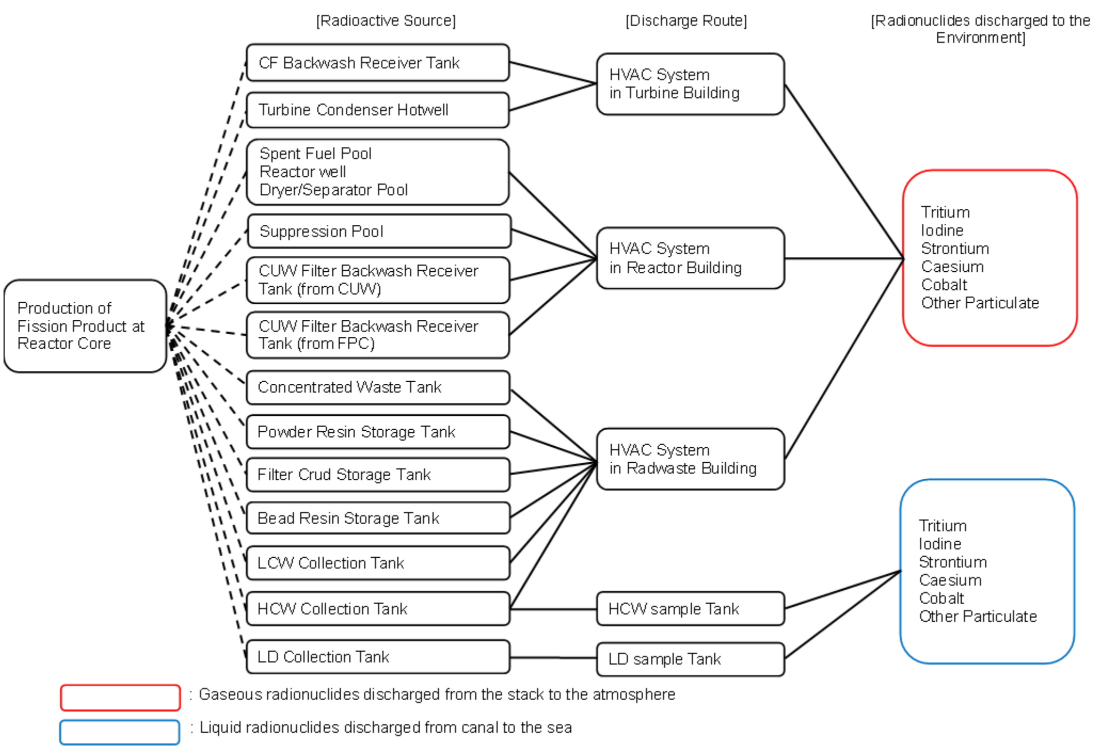

Figure 6-2: Radioactive Discharge Routes to the Environment during Outage

7. Discharge Assessments

7.1. Gaseous Discharges

Monthly discharges and annual discharges are calculated in line with the requirements of the P&ID. As outlined in section 6, the gaseous discharges from the OG, HVAC and TGS systems will vary depending on the operating mode of the reactor. For the purposes of the quantification of annual releases to the environment, discharges from the OG, HVAC and TGS systems are set with 12 months of operation which includes a one month period of planned maintenance/outage [Ref-10].

No measurement data is readily available, therefore the monthly predicted gaseous discharges presented in sections 7.1.1, 7.1.2 and 7.1.3 are based on theoretical calculation of the production of radioisotopes in water, subsequent partitioning and transfer to steam and the performance of the process systems in the relevant discharge streams. As a result, the figures showing the predicted monthly discharge are somewhat idealised. The figures show fixed discharge rates occurring during power operations and during outage with instantaneous variation between the discharge rate associated with one operational mode and the following operating mode resulting in a regular rectangular profile. In practice the discharge rate will vary around the calculated values as illustrated in Figure 7.1-1.

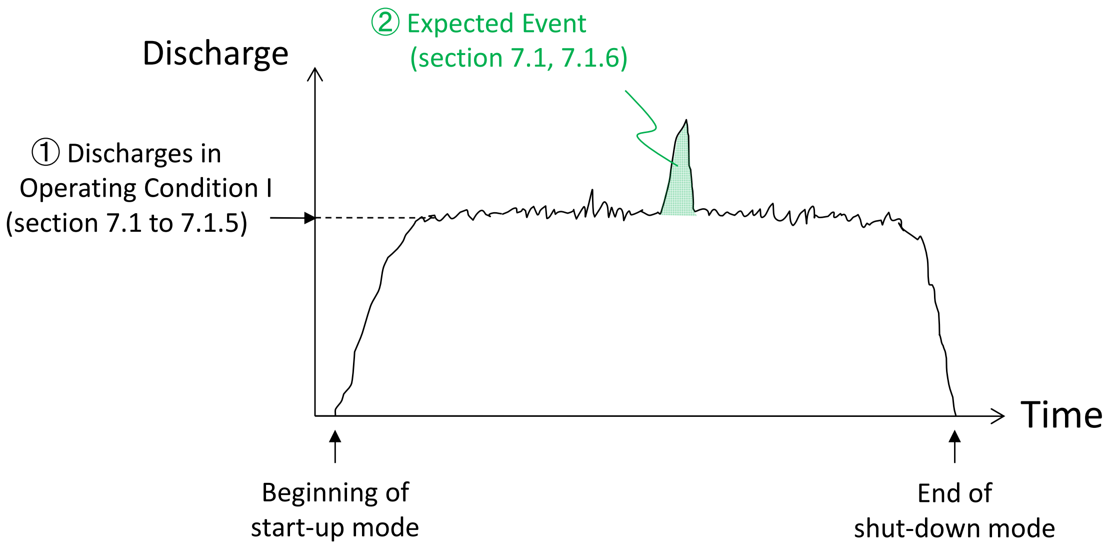

Figure 7.1-1: Illustrative Profile of a Realistic Gaseous Discharge Including a Contribution due to an Expected Event

As can be seen from Figure 7.1-1 the variation in discharge rate when the reactor changes operational mode is not instantaneous but instead occurs over a period of time (in the case of start-ups and shutdowns, this is typically about a day). Similarly when the system is at a steady state there will be small fluctuations in the discharge rate.

The gaseous discharges from the OG, HVAC and TGS systems will vary depending on the operating mode of the reactor. It is worth noting that, whilst temporary increases in iodine and CP activity concentrations are observed in the reactor water circuit during major changes of state of a reactor system, e.g. at shutdown, these so called “iodine spikes” and “crud bursts” have little impact on gaseous discharges to the environment, due to the reactor water clean-up and discharge mitigation measures. Due account of these events is taken by the use of the bounding Source Term in each scenario, as detailed in Table 7.1-1.

Table 7.1-1 summarises the operating modes (such as Power operation and Shutdown) and types of source term (BE, LCO, CA) used for each gaseous discharge route that contributes to the Main Stack, along with a brief justification for the use of each type of ST and the system location point at which the activity is calculated.

Table 7.1-1: Summary of Contributing Discharge Routes, Operating Modes and Source Terms used to Calculate Gaseous Discharges (1/4)

N.B. The Operating Mode definitions, as used in Table 7.1-2, 7.1-4, 7.1-6 and 7.1-9, are: (1) - Power operation, (2) – Start-up, (3) – Shutdown, (4) – Outage, (5) - Expected Event (Fuel pin failure).

Discharge Point to the Environment	Discharge Building	Discharge Route	PrST calculation point	Operating modes considered and type of ST used	Remarks
Stack	Turbine Building (T/B)	OG system	OG system outlet	(1)(2)(3) → **Power Operation (BE)** (5) → **Power Operation (LCO)**	(1)(2)(3): Power Operation (BE) is used during power operation discharges. Start up and Shut down are bounded by the Power Operation (BE), i.e. highest value in (1)(2)(3) is used. N.B. Startup and shutdown periods are normally completed within ~ 1 day. By comparison power operation lasts for approximately 517 days. Therefore, the total activity of considering all three operating modes together (as required for deriving annual discharges) is dominated by the Power Operation source term meaning use of the Power Operation (BE) source term is justified. Furthermore, FP will not be discharged during outages, therefore the Outage source term is not required to capture such activity. (5): The Power Operation (LCO) source term will be used to incorporate the discharge contribution of Expected Event (fuel pin failure).

Table 7.1-1: Summary of Contributing Discharge Routes, Operating Modes and Source Terms used to Calculate Gaseous Discharges (2/4)

Discharge Point to the Environment	Discharge Building	Discharge Route	PrST calculation point	Operating modes considered and type of ST used	Remarks
Stack	Turbine Building (T/B)	HVAC system	Condenser hotwell	(4) → **Outage (BE)**	(4): This discharge occurs when the manhole cover on the condenser hotwell is opened during outage. The outage (BE) source term is therefore used in the discharge calculation. The effects of crud burst and iodine spike during shutdown are considered in the outage (BE) source term.
			CF Backwash Receiver Tank	(1)(2)(3)(4) → **CA (BE)**	(1)(2)(3)(4): The fluctuations in activity concentration in the tank over the operational cycle are captured in the CA (BE) source term (i.e. the effects of crud burst and iodine spike during shutdown are incorporated).
		TGS system	Condensate Storage Tank (CST)	(1)(2)(3) → **Power Operation (BE)**	(1)(2)(3): The TGS system is operated in conjunction with reactor operation. Therefore, the Power Operation (BE) is used in (1)(2)(3) as per the operation of the OG system.
	Reactor Building (R/B)	HVAC system	Spent Fuel Storage Pool (SFP)	(1)(2)(3) → **Power Operation (BE)** (4) → **Outage (BE)**	(1)(2)(3): The concentration of the SFP is the same between phase 1 to 3 (start-up, shutdown and power operation) therefore the Power Operation (BE) source term is used. The activity level used is derived from the concentration of reactor water. (4): The outage (BE) is used to set the concentration of the SFP during outage. However, The concentration is the same with the Power Operation (BE) of the SFP in PrST because there is no significant fluctuation

Table 7.1-1: Summary of Contributing Discharge Routes, Operating Modes and Source Terms used to Calculate Gaseous Discharges (3/4)

Discharge Point to the Environment	Discharge Building	Discharge Route	PrST calculation point	Operating modes considered and type of ST used	Remarks
Stack	Reactor Building (R/B)	HVAC system	Reactor Well (same as SFP)	(4) → **Outage (BE)**	(4): The Reactor Well is connected to SFP during outage. Therefore, the same source term as that used for the SFP is used.
			D/S pool (same as SFP)	(4) → **Outage (BE)**	(4): D/S pool is connected to SFP during outage. Therefore, the same source term as that used for the SFP is used.
			Suppression Pool	(4) → **Outage (BE)**	(4): This discharge occurs during an outage only because the suppression pool is ventilated during outage for worker entry. The Outage (BE) source term is therefore used in the discharge calculation.
			CUW Filter Backwash Receiver Tank	(1)(2)(3)(4) → **CA (BE)**	(1)(2)(3)(4): The fluctuations in activity concentration in the tank over the operational cycle are captured in the CA (BE) source term (i.e. the effects of crud burst and iodine spike during shutdown are incorporated).

Table 7.1-1: Summary of Contributing Discharge Routes, Operating Modes and Source Terms used to Calculate Gaseous Discharges (4/4)

Discharge Point to the Environment	Discharge Building	Discharge Route	PrST calculation point	Operating modes considered and type of ST used	Remarks
Stack	Radwaste Building (Rw/B)	HVAC system	Low Chemical Impurities Water System (LCW) Collection Tank	(1)(2)(3)(4) → **CA (BE)**	(1)(2)(3)(4): The fluctuations in activity concentration in each tank over the operational cycle are captured in the CA (BE) source term for each tank (i.e. the effects of crud burst and iodine spike during shutdown are incorporated).
			High Chemical Impurities Water System (HCW) Collection Tank		
			Powder Resin Storage Tank		
			Filter Crud Storage Tank		
			Bead Resin Storage Tank		
			Concentrated Waste Tank		

7.1.1. Monthly Gaseous Discharges – OG System

The radioactive discharges from the OG system arise as volatile non-condensable gases such as noble gases and C-14 are removed from the reactor water circuit at the Steam Jet Air Ejector (SJAE) and are discharged via the OG system. Condensable gases (such as tritiated water) and particulates remain in the reactor water circuit and so are not expected in the discharges in the OG system.

A schematic diagram showing the route for gaseous discharge via the OG system is shown in Figure 7.1-2, and the route for the gaseous discharges via OG system during reactor start-up, power operation and shutdown, is described in STEPs 1 to 4 below, with the 'Item' numbers corresponding to the numbers shown in Figure 7.1-2.

STEP1: Radionuclides in the reactor water are generated in the reactor core. *Item 1*

STEP2: Radionuclides in the reactor water migrate into the main steam. *Item 2*

STEP3: Radionuclides that migrated into the main steam are cooled by the main condenser, and non-condensable gaseous nuclides (noble-gases, iodine and C-14) are extracted into the OG system. *Item 3*

STEP4: Gaseous nuclides (noble-gases, iodine and C-14) that migrated into the OG system are held up in the OG system and then discharged from the stack (almost all the iodine is retained in the OG system). *Items 4 and 5*

During outage, the turbine and condensers are isolated from the reactor by closure of the Main Steam Isolation Valves. The condensers are let up to ambient atmospheric pressure. Therefore no discharge is made through the OG system during outage.

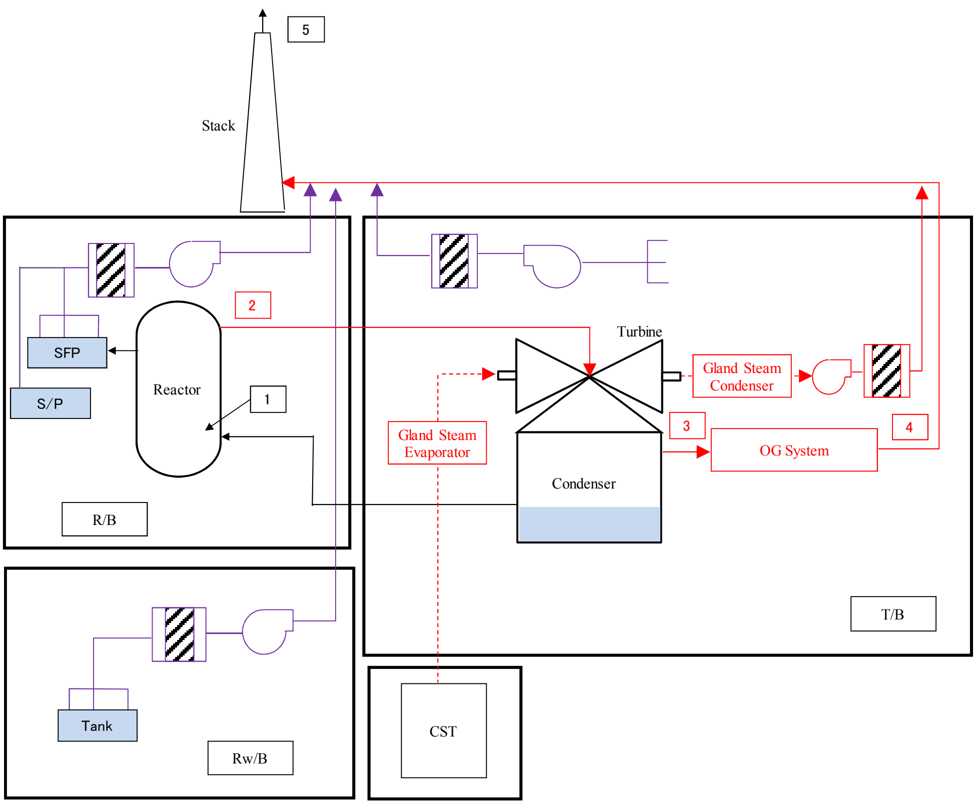

Figure 7.1-2: Schematic of Gaseous Discharge for OG System

The monthly discharges of radionuclides from the OG system have been calculated based on the transport of radioactivity through the reactor, main steam line and the OG system, shown in Figure 7.1-2 and Figure B1 of Appendix B. The calculations take appropriate account of flow rates, partitioning between water and steam, separation into the OG system, removal in the OG system condenser and carbon delay beds. As the OG system is extremely effective in removing tritiated water, iodine and particulates from the gas stream, only negligibly small amounts of the associated radionuclides (i.e. H-3, iodine and radionuclides associated with particulate matter) are present in the discharged gas, and therefore for the purposes of this assessment the discharges of these radionuclides are considered to be zero.

The concentration of radionuclides in the discharged gas is based on PrST calculation point OG-7-PO-BE and shown in Table 7.1-2 (a diagram showing the location of the calculation point is presented in Appendix B). The resulting monthly radioactive discharges from the OG system have been determined using the data presented in [Ref-10] and are presented in Table 7.1-3.

Table 7.1-2: Radioactive Concentration in the OG System Exhaust

Radionuclides	Concentration in Reactor Steam (Bq/cm^3) PrST: OG-7-PO-BE
Ar-41	5.1E+00
Kr-85	2.9E-04
Kr-85m	6.7E-03
Kr-87	6.7E-09
Kr-88	5.0E-04
Kr-89	0.0E+00
Xe-131m	4.0E-04
Xe-133	2.9E-02
Xe-133m	4.8E-06
Xe-135	5.0E-23
Xe-135m	0.0E+00
Xe-137	0.0E+00
Xe-138	0.0E+00
C-14	2.6E+00

Table 7.1-3: Monthly Discharge from the OG System

Radionuclides	Monthly Discharges during Power Operation (Bq/month)	Monthly Discharges during Outage (Bq/month)
Ar-41	1.5E+11	0.0E+00
Kr-85	8.5E+06	0.0E+00
Kr-85m	1.9E+08	0.0E+00
Kr-87	1.9E+02	0.0E+00
Kr-88	1.5E+07	0.0E+00
Kr-89	0.0E+00	0.0E+00
Xe-131m	1.2E+07	0.0E+00
Xe-133	8.4E+08	0.0E+00
Xe-133m	1.4E+05	0.0E+00
Xe-135	1.4E-12	0.0E+00
Xe-135m	0.0E+00	0.0E+00
Xe-137	0.0E+00	0.0E+00
Xe-138	0.0E+00	0.0E+00
C-14	7.6E+10	0.0E+00

N.B. No discharges of H-3, iodine or particulates are expected from the OG system [Ref-10].

If the discharges are grouped into: (i) Noble gases excluding Ar-41; (ii) Ar-41; and (iii) C-14 then the monthly discharges to atmosphere will occur as illustrated in Figures 7.1-3, 7.1-4 and 7.1-5, respectively (see comments under Figure 7.1-1 regarding the “idealised” nature of these graphs).

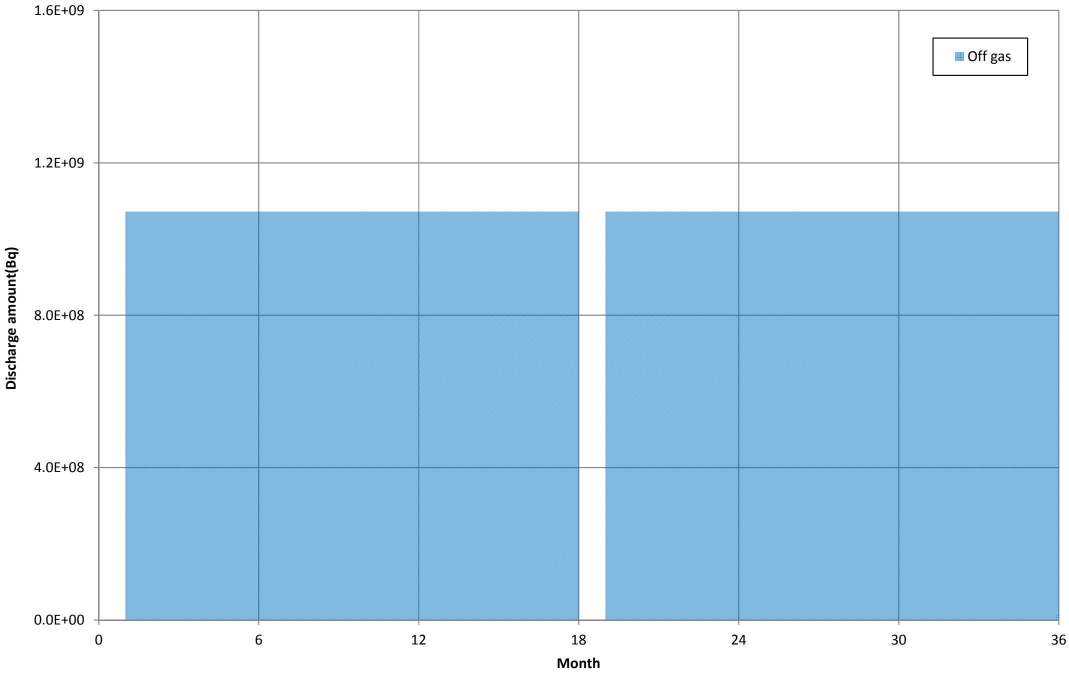

Figure 7.1-3: Predicted Monthly Discharge of Noble Gases (excluding Ar-41) from the OG System

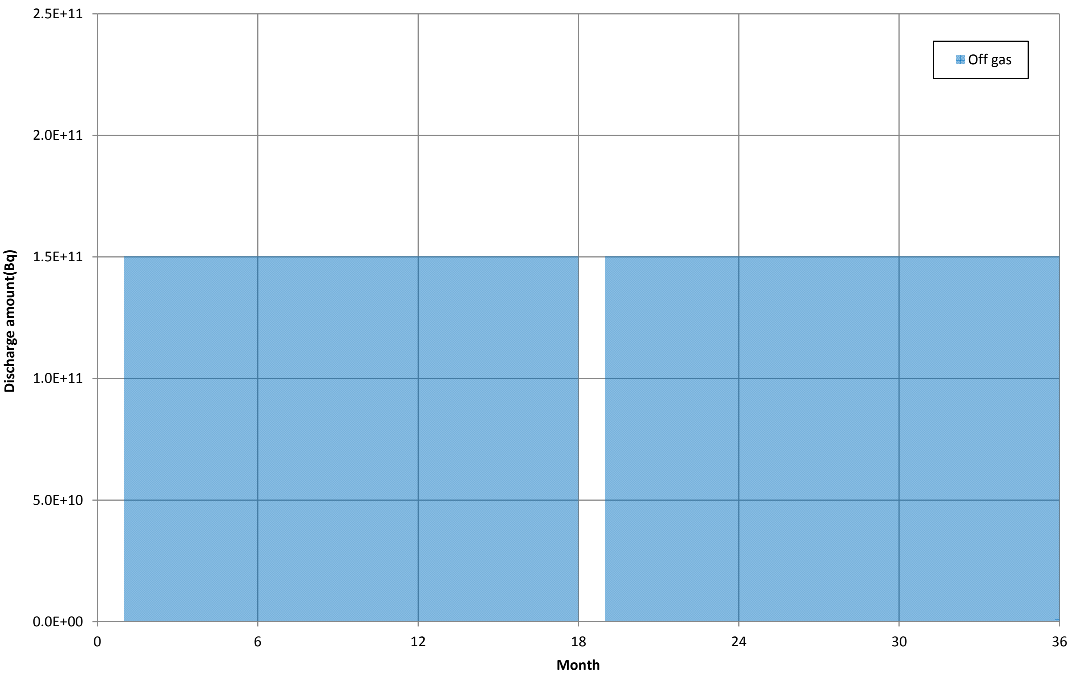

Figure 7.1-4: Predicted Monthly Ar-41 Discharge of from the OG System

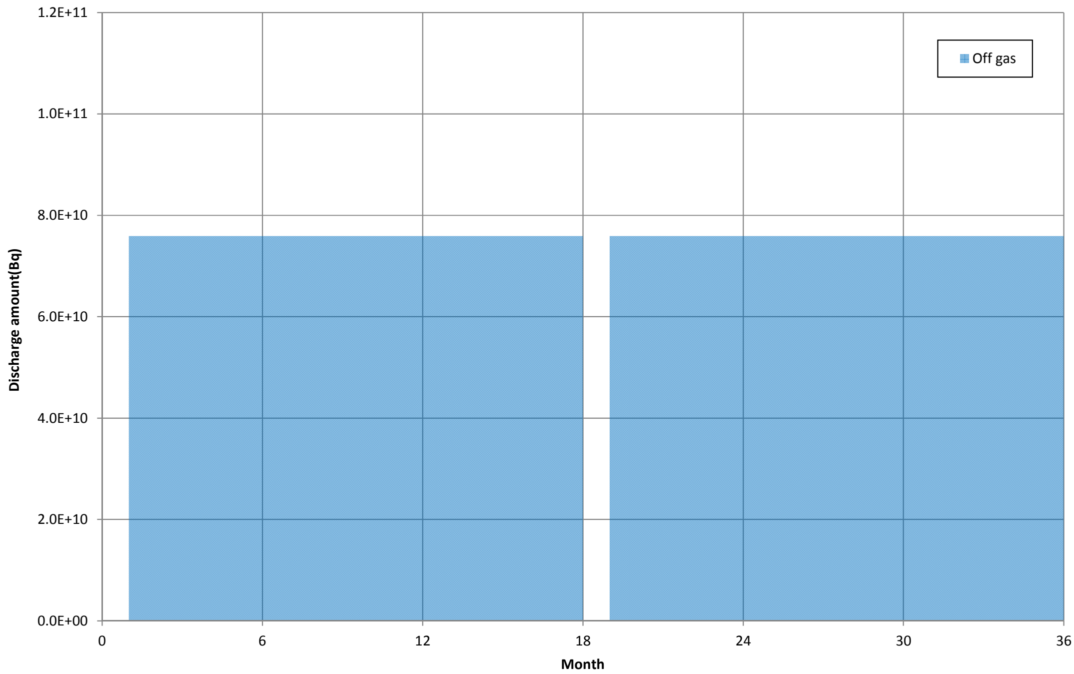

Figure 7.1-5: Predicted Monthly C-14 Discharge from the OG System

7.1.2. Monthly Gaseous Discharges – HVAC System

The radioactive discharges from the HVAC system are aggregated from the contributing plant systems serving the Reactor, Radwaste and Turbine Buildings, as outlined in Figure 6-1 and 6-2 and illustrated in Figure 7.1-6. It should be noted that the HVAC system discharge also includes the tank vents/extracts from the various Rw/B tanks (the extract from which joins the Rw/B HVAC system ducting before it exits the Rw/B).

During start-up, power operation and shutdown, discharge routes for gaseous discharges made via the HVAC system are described in STEP1 below (the 'Item' numbers correspond with the numbers in Figure 7.1-6).

STEP1: H-3, iodine and particulate nuclides migrate into the vapour from the SFP and from the waste system tanks in each building, and are discharged from the stack via the HVAC system. *Item 1, 2, 3 and 4*

In addition to the discharge routes in the STEP1, the additional discharge routes for gaseous discharges during outage are described in STEP 2 and 3 below.

STEP2: H-3, iodine and particulate nuclides migrate into the vapour from the reactor well and dryer/separator (D/S) pool after the Reactor Pressure Vessel (RPV) is opened for fuel change during outage. Also, H-3, iodine and particulate nuclides migrate into the vapour from suppression pool during outage as the air in the S/C is ventilated to allow workers to enter the enclosure. The associated radionuclides are discharged from the stack via the HVAC system. *Items 1 and 4*

STEP3: H-3, iodine and particulate nuclides migrate into the vapour released from the main condenser hot-well after the turbine casing is opened for maintenance during an outage, and are discharged from the stack via the HVAC system. *Items 2 and 4*

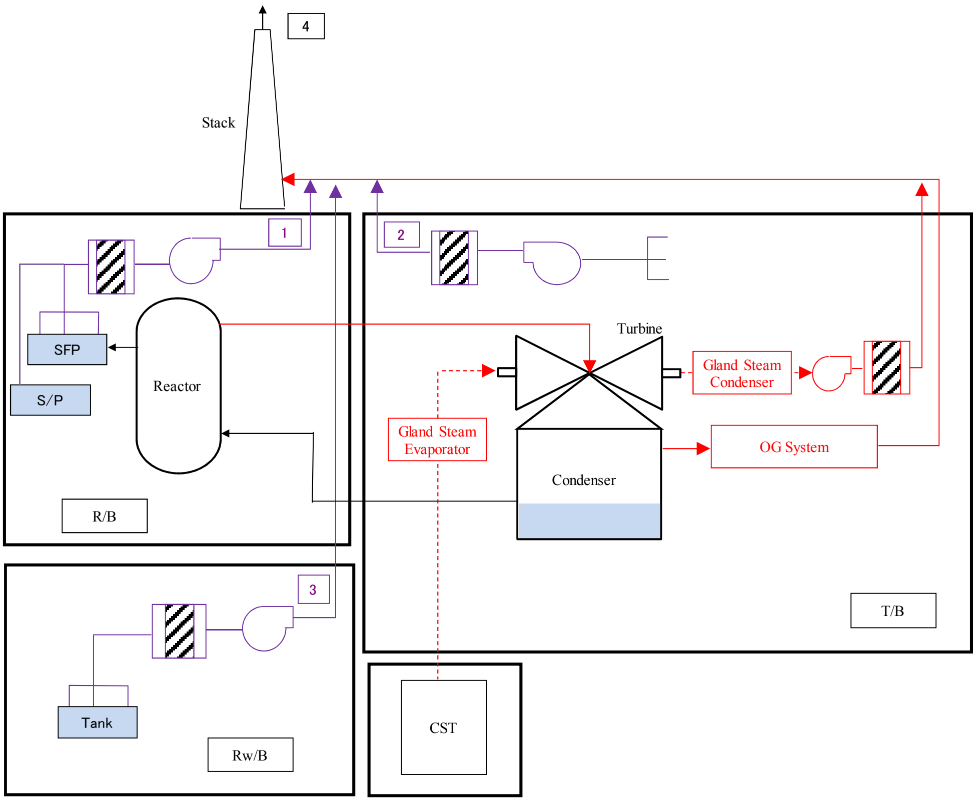

Figure 7.1-6: Schematic of Gaseous Discharges for the HVAC system

H-3, iodine and particulate nuclides migrate from the water in each plant system into the air. Particulates are removed from the air by using High Efficiency Particulate Air (HEPA) filters, and the filtered air is then discharged from the HVAC system via the stack.

Volatile non-condensable gases are removed from the reactor water/steam circuit at the SJAE and are discharged via the OG system as outlined in section 7.1.1. As a result, noble gases and C-14 are typically not present in the systems that are served by the HVAC system and so very low concentrations would be expected in the HVAC system discharge. Therefore, for the purposes of this assessment the discharge of noble gases and C-14 from the HVAC system is assumed to be zero.

The monthly discharges of radionuclides from the HVAC system have been calculated based on the radioactivity present in the SFP, reactor well, D/S pool, suppression pool, condenser hot well and various tanks containing reactor water or supernatant or washings from liquid waste systems. The calculations take account of the transition rates of radioactivity through these systems, including partitioning between water and air, water temperature, evaporation rates, number of contributing vessels and removal by abatement systems.

The concentrations of radioactive species were taken from the following PrST calculation points and shown in Table 7.1-4 (the locations of these calculation points are described in Appendix B).

SFP	FPC-1-PO-BE (power operation)
SFP	FPC-1-outage-BE (outage)
Reactor well	FPC-1-outage-BE (outage)
D/S pool	FPC-1-outage-BE (outage)

Suppression Pool	SP-1-outage-BE (outage)
CUW Filter Backwash Receiver Tank (from CUW)	SS-2-CA-BE
CUW Filter Backwash Receiver Tank (from FPC)	SS-3-CA-BE
Turbine Condenser Hotwell	C-1-outage-BE (outage)
CF Backwash Receiver Tank	CPS-3-CA-BE
LCW Collection Tank	LCW-1-CA-BE
HCW Collection Tank	HCW-1-CA-BE
Powder Resin Storage Tank	SS-10-CA-BE (CUW decant water)
Filter Crud Storage Tank	SS-13-CA-BE (LCW decant water)
Bead Resin Storage Tank	SS-14-CA-BE (CD decant water)
Concentrated Waste Tank	HCW-2-CA-BE

The calculated monthly discharges from the HVAC system are presented in Table 7.1-5. In Table 7.1-5, the radionuclides have been grouped into particulates (comprising the radionuclides highlighted in brown), iodine (highlighted in purple) and H-3. The monthly discharges are shown for each operating mode.

Discharges to the environment from the R/B arise from the SFP, the CUW filter backwash receiver tank (from CUW) and the CUW filter backwash receiver tank (from FPC) during all operating modes and with additional arisings from the reactor well, D/S pool and suppression pool during outage.

Discharges to the environment from the CUW filter backwash receiver tank occur over all operating modes. The discharges to the HVAC from the CUW filter back wash receiver tank are not continuous but occur each time the CUW filter is backwashed. The CUW filter backwash receiver tank is used both in the CUW process and the FPC process. The durations of monthly discharge are 38 hours per month for CUW process and 12 hours per month for FPC process based on each one's backwash frequency.

Discharges to the environment from the T/B are from the CF Backwash Receiver Tank during all four operating modes and from the turbine condenser hotwell during outage.

Discharges to the environment from the CF Backwash Receiver Tank occur over all operating modes. The discharges to the HVAC from the CF Backwash Receiver Tank are not continuous but occur each time the CF is backwashed. The duration of monthly discharge is 74 hours for the CF filter based on CF backwash frequency.

It is considered that the discharges to the HVAC from the backwash receiver tanks do not need to be separately considered from the other monthly discharges to the environment from the HVAC as the discharges from the backwash tanks are frequent.

Discharges to the environment from the Rw/B arise from the LCW collection tank, the HCW collection tank, the Sludge Storage (S/S) tank and the Spent Resin Storage Tank during all operating modes.

The monthly discharges of particulate, iodine and H-3 from the HVAC system from each of the contributing buildings are depicted in Figures 7.1-7, 7.1-8 and 7.1-9 respectively (see comments under Figure 7.1-1 regarding the "idealised" nature of these graphs).

Table 7.1-4: Radionuclide Concentrations in the Systems that are Served by the HVAC System

	R/B							T/B		Rw/B					
	SFP	SFP	Reactor well	D/S pool	Suppression pool	CUW Filter Backwash Receiver Tank (from CUW)	CUW Filter Backwash Receiver Tank (from FPC)	Turbine Condenser Hotwell	CF Backwash Receiver Tank	LCW Collection Tank	HCW Collection Tank	Powder Resin Storage Tank	Filter Crud Storage Tank	Bead Resin Storage Tank	Concentrated Waste Tank
Operating mode*	(1)(2)(3)	(4)	(4)	(4)	(4)	(1)(2)(3)(4)	(1)(2)(3)(4)	(4)	(1)(2)(3)(4)	(1)(2)(3)(4)					
PrST calculation point	FPC-1-PO-BE (Power Operation)	FPC-1-outage-BE	FPC-1-outage-BE	FPC-1-outage-BE	SP-1-outage-BE	SS-2-CA-BE	SS-3-CA-BE	C-1-outage-BE	CPS-3-CA-BE	LCW-1-CA-BE	HCW-1-CA-BE	SS-10-CA-BE (CUW decant water)	SS-13-CA-BE (LCW decant water)	SS-14-CA-BE (CD decant water)	HCW-2-CA-BE
Radionuclides	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3	Bq/m^3
Cr-51	1.1E+06	1.1E+06	Same as SFP((4)Outage)		9.3E+05	1.9E+11	1.9E+10	5.5E+05	1.0E+08	1.6E+07	1.5E+04	1.5E+08	6.9E+08	1.0E+05	4.8E+06
Mn-54	3.6E+05	3.6E+05			2.6E+05	9.0E+10	1.3E+10	1.5E+05	5.5E+07	9.5E+06	1.6E+04	7.1E+07	7.6E+08	3.2E+05	3.2E+07
Co-58	1.1E+06	1.1E+06			1.9E+05	1.7E+11	2.9E+10	1.1E+05	7.4E+07	1.3E+07	2.2E+04	1.4E+08	6.2E+08	2.9E+05	1.3E+07
Fe-59	1.2E+05	1.2E+05			9.0E+04	2.4E+10	2.9E+09	5.3E+04	2.1E+07	2.5E+06	1.6E+03	1.9E+07	1.6E+08	1.0E+04	6.8E+05
Co-60	5.4E+05	5.4E+05			4.4E+05	1.5E+11	2.0E+10	2.6E+05	7.1E+07	1.8E+07	7.0E+04	1.2E+08	1.5E+09	1.8E+06	2.9E+08
Zn-65	2.8E+05	2.8E+05			1.7E+05	6.5E+10	9.5E+09	9.8E+04	2.0E+07	4.7E+06	1.4E+04	5.1E+07	2.9E+08	2.9E+05	2.4E+07
Sr-89	5.5E+01	5.5E+01			5.4E+02	1.4E+10	1.3E+06	3.2E+02	5.5E+06	9.1E+05	3.8E+03	1.1E+07	3.0E+07	5.1E+04	1.8E+06
Sr-90	2.3E+00	2.3E+00			2.4E+01	7.4E+08	8.8E+04	1.4E+01	1.7E+05	5.7E+04	1.8E+03	5.8E+05	3.0E+06	4.8E+04	8.6E+06
Zr-95	3.2E+03	3.2E+03			3.8E+03	1.9E+10	8.6E+07	2.2E+03	5.1E+06	1.1E+06	5.7E+03	1.5E+07	3.3E+07	8.5E+04	3.1E+06
Nb-95	3.4E+03	3.4E+03			4.2E+03	4.1E+10	1.9E+08	2.4E+03	1.1E+07	2.4E+06	1.3E+04	3.3E+07	7.3E+07	1.9E+05	6.9E+06
Ag-110m	1.1E+02	1.1E+02			1.1E+02	3.1E+07	3.8E+06	6.3E+01	3.1E+04	4.3E+03	1.8E+00	2.4E+04	4.0E+05	1.7E+01	3.1E+03
Sb-122	2.5E+03	2.5E+03			2.4E+03	7.1E+07	5.3E+06	1.4E+03	3.2E+04	1.2E+04	1.3E+01	5.6E+04	8.4E+04	2.9E+01	2.8E+03
Sb-124	1.1E+03	1.1E+03			1.8E+03	1.3E+10	2.9E+07	1.0E+03	1.1E+06	6.2E+05	2.2E+03	1.0E+07	8.5E+06	3.0E+04	1.1E+06
Sb-125	9.3E+01	9.3E+01			1.6E+02	1.6E+09	3.5E+06	9.2E+01	4.8E+05	1.3E+05	1.2E+03	1.3E+06	7.7E+06	3.1E+04	4.4E+06
Cs-134	5.6E+02	5.6E+02			6.9E+02	2.2E+09	2.1E+07	4.0E+02	1.2E+06	2.3E+05	2.5E+03	1.7E+06	1.6E+07	6.4E+04	8.2E+06
Cs-137	6.6E+00	6.6E+00			6.9E+01	1.1E+09	2.5E+05	4.1E+01	7.4E+05	1.7E+05	2.7E+03	8.9E+05	1.3E+07	7.1E+04	1.3E+07
Ba-140	7.6E+01	7.6E+01			8.0E+02	1.1E+10	7.4E+05	7.0E+02	3.2E+06	7.4E+05	2.7E+03	8.6E+06	6.3E+06	2.1E+04	6.1E+05
La-140	1.3E+02	1.3E+02			1.2E+03	1.3E+10	8.6E+05	6.9E+02	3.1E+06	8.5E+05	3.1E+03	1.0E+07	7.2E+06	2.5E+04	7.1E+05
Ce-141	5.0E+02	5.0E+02			1.1E+03	1.6E+10	9.8E+06	6.6E+02	6.9E+06	1.1E+06	4.0E+03	1.3E+07	3.1E+07	4.4E+04	1.4E+06
Ce-144	4.9E+01	4.9E+01			5.1E+02	1.5E+10	1.7E+06	3.0E+02	4.2E+06	9.5E+05	1.1E+04	1.2E+07	4.0E+07	2.4E+05	2.0E+07
Pu-238	3.2E-06	3.2E-06			3.4E-05	2.3E+03	1.2E-01	2.0E-05	4.0E+00	7.7E-01	7.0E-03	1.8E+00	7.1E+01	1.9E-01	3.4E+01
Pu-239	4.0E-07	4.0E-07			4.3E-06	2.9E+02	1.5E-02	2.5E-06	5.1E-01	9.9E-02	9.1E-04	2.3E-01	9.1E+00	2.4E-02	4.5E+00
Pu-240	6.5E-07	6.5E-07			6.9E-06	4.7E+02	2.5E-02	4.0E-06	8.1E-01	1.6E-01	1.5E-03	3.7E-01	1.5E+01	3.9E-02	7.2E+00
Am-241	2.2E-07	2.2E-07			2.3E-06	1.6E+02	8.3E-03	1.3E-06	2.7E-01	5.3E-02	4.9E-04	1.2E-01	4.9E+00	1.3E-02	2.4E+00
Cm-242	3.1E-04	3.1E-04			3.3E-03	2.1E+05	1.0E+01	1.9E-03	3.7E+02	3.5E+01	1.2E-01	1.6E+02	2.7E+03	2.4E+00	1.5E+02
Cm-243	1.8E-08	1.8E-08			1.9E-07	1.3E+01	6.7E-04	1.1E-07	2.2E-02	4.2E-03	3.7E-05	1.0E-02	3.8E-01	9.8E-04	1.8E-01
Cm-244	2.2E-06	2.2E-06			2.4E-05	1.6E+03	8.5E-02	1.4E-05	2.8E+00	5.2E-01	4.6E-03	1.3E+00	4.8E+01	1.2E-01	2.2E+01
I-131	2.2E+01	2.2E+01			2.3E+02	8.0E+09	1.4E+05	1.1E+04	0.0E+00	4.4E+06	6.2E+04	1.3E+08	2.8E-01	1.2E+07	1.3E+07
I-132	1.2E+02	1.2E+02			1.3E+03	3.0E+09	1.8E+05	6.4E+04	0.0E+00	4.8E+05	7.2E+02	2.4E+06	7.4E-04	4.8E+03	2.3E+05
I-133	7.0E+01	7.0E+01			7.4E+02	1.3E+09	4.7E+04	3.6E+04	0.0E+00	1.2E+06	4.0E+03	2.1E+07	1.4E-02	1.9E+06	8.4E+05
I-135	9.7E+01	9.7E+01			1.0E+03	3.4E+08	2.1E+04	5.0E+04	0.0E+00	6.2E+05	1.5E+03	5.4E+06	3.6E-03	5.0E+05	3.0E+05
H-3	3.5E+08	3.5E+08			3.5E+08	0.0E+00	0.0E+00	3.5E+08	0.0E+00	3.5E+08	3.5E+08	3.5E+08	3.5E+08	3.5E+08	0.0E+00

*: (1) Power operation, (2) Start-up, (3) Shutdown, (4) Outage

Table 7.1-5: Monthly Gaseous Discharges via the HVAC System

	R/B							T/B		Rw/B					
	SFP	SFP	Reactor well	D/S pool	Suppression pool	CUW Filter Backwash Receiver Tank (from CUW)	CUW Filter Backwash Receiver Tank (from FPC)	Turbine Condenser Hotwell	CF Backwash Receiver Tank	LCW Collection Tank	HCW Collection Tank	Powder Resin Storage Tank	Filter Crud Storage Tank	Bead Resin Storage Tank	Concentrated Waste Tank
Operating mode*	(1)(2)(3)	(4)	(4)	(4)	(4)	(1)(2)(3)(4)	(1)(2)(3)(4)	(4)	(1)(2)(3)(4)	(1)(2)(3)(4)					
Duration of Monthly Discharge(h)	730	730	730	730	730	38	12	730	74	730	730	730	730	730	730
Radionuclides	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month	Bq/month
Cr-51	5.4E+00	5.4E+00	3.9E+00	3.8E+00	7.4E+00	4.3E+02	1.4E+01	4.3E-01	1.2E+00	2.5E+01	4.0E-03	2.0E+01	1.8E+02	8.1E-02	2.1E-01
Mn-54	1.8E+00	1.8E+00	1.3E+00	1.3E+00	2.0E+00	2.0E+02	9.2E+00	1.2E-01	6.4E-01	1.5E+01	4.2E-03	9.3E+00	2.0E+02	2.6E-01	1.4E+00
Co-58	5.4E+00	5.4E+00	3.9E+00	3.8E+00	1.5E+00	3.9E+02	2.1E+01	9.0E-02	8.5E-01	2.0E+01	5.7E-03	1.8E+01	1.6E+02	2.3E-01	5.7E-01
Fe-59	6.2E-01	6.2E-01	4.4E-01	4.3E-01	7.1E-01	5.5E+01	2.1E+00	4.2E-02	2.4E-01	4.0E+00	4.2E-04	2.5E+00	4.2E+01	8.2E-03	3.0E-02
Co-60	2.7E+00	2.7E+00	2.0E+00	1.9E+00	3.5E+00	3.3E+02	1.5E+01	2.1E-01	8.2E-01	2.8E+01	1.8E-02	1.5E+01	3.9E+02	1.4E+00	1.3E+01
Zn-65	1.4E+00	1.4E+00	1.0E+00	9.7E-01	1.3E+00	1.5E+02	6.9E+00	7.8E-02	2.4E-01	7.5E+00	3.7E-03	6.7E+00	7.6E+01	2.3E-01	1.0E+00
Sr-89	2.8E-04	2.8E-04	2.0E-04	1.9E-04	4.3E-03	3.1E+01	9.6E-04	2.5E-04	6.4E-02	1.4E+00	1.0E-03	1.4E+00	7.9E+00	4.0E-02	7.7E-02
Sr-90	1.2E-05	1.2E-05	8.4E-06	8.1E-06	1.9E-04	1.7E+00	6.3E-05	1.1E-05	1.9E-03	8.9E-02	4.7E-04	7.6E-02	8.0E-01	3.8E-02	3.8E-01
Zr-95	1.6E-02	1.6E-02	1.2E-02	1.1E-02	3.0E-02	4.3E+01	6.2E-02	1.8E-03	5.9E-02	1.7E+00	1.5E-03	1.9E+00	8.7E+00	6.7E-02	1.4E-01
Nb-95	1.7E-02	1.7E-02	1.3E-02	1.2E-02	3.3E-02	9.4E+01	1.4E-01	1.9E-03	1.3E-01	3.8E+00	3.3E-03	4.3E+00	1.9E+01	1.5E-01	3.0E-01
Ag-110m	5.6E-04	5.6E-04	4.0E-04	3.9E-04	8.5E-04	7.0E-02	2.7E-03	5.0E-05	3.6E-04	6.9E-03	4.7E-07	3.2E-03	1.1E-01	1.4E-05	1.3E-04
Sb-122	1.2E-02	1.2E-02	8.9E-03	8.7E-03	1.9E-02	1.6E-01	3.8E-03	1.1E-03	3.7E-04	2.0E-02	3.5E-06	7.3E-03	2.2E-02	2.3E-05	1.2E-04
Sb-124	5.7E-03	5.7E-03	4.1E-03	3.9E-03	1.4E-02	3.0E+01	2.1E-02	8.2E-04	1.3E-02	9.8E-01	5.8E-04	1.4E+00	2.2E+00	2.4E-02	5.0E-02
Sb-125	4.7E-04	4.7E-04	3.4E-04	3.3E-04	1.2E-03	3.7E+00	2.5E-03	7.3E-05	5.5E-03	2.0E-01	3.2E-04	1.7E-01	2.0E+00	2.4E-02	1.9E-01
Cs-134	2.8E-03	2.8E-03	2.0E-03	2.0E-03	5.4E-03	5.0E+00	1.5E-02	3.2E-04	1.4E-02	3.6E-01	6.6E-04	2.3E-01	4.2E+00	5.0E-02	3.6E-01
Cs-137	3.3E-05	3.3E-05	2.4E-05	2.3E-05	5.5E-04	2.6E+00	1.8E-04	3.3E-05	8.5E-03	2.7E-01	7.0E-04	1.2E-01	3.5E+00	5.6E-02	5.6E-01
Ba-140	3.8E-04	3.8E-04	2.7E-04	2.7E-04	6.3E-03	2.5E+01	5.3E-04	5.6E-04	3.7E-02	1.2E+00	7.1E-04	1.1E+00	1.6E+00	1.7E-02	2.7E-02
La-140	6.6E-04	6.6E-04	4.7E-04	4.6E-04	9.3E-03	2.9E+01	6.2E-04	5.5E-04	3.5E-02	1.3E+00	8.1E-04	1.3E+00	1.9E+00	1.9E-02	3.1E-02
Ce-141	2.5E-03	2.5E-03	1.8E-03	1.7E-03	8.9E-03	3.6E+01	7.1E-03	5.3E-04	8.0E-02	1.7E+00	1.1E-03	1.7E+00	8.1E+00	3.5E-02	6.0E-02
Ce-144	2.5E-04	2.5E-04	1.8E-04	1.7E-04	4.1E-03	3.4E+01	1.2E-03	2.4E-04	4.8E-02	1.5E+00	2.8E-03	1.6E+00	1.1E+01	1.9E-01	8.8E-01
Pu-238	1.6E-11	1.6E-11	1.2E-11	1.1E-11	2.6E-10	5.2E-06	8.7E-11	1.6E-11	4.6E-08	1.2E-06	1.8E-09	2.4E-07	1.9E-05	1.5E-07	1.5E-06
Pu-239	2.0E-12	2.0E-12	1.5E-12	1.4E-12	3.4E-11	6.6E-07	1.1E-11	2.0E-12	5.8E-09	1.6E-07	2.4E-10	3.0E-08	2.4E-06	1.9E-08	2.0E-07
Pu-240	3.3E-12	3.3E-12	2.4E-12	2.3E-12	5.4E-11	1.1E-06	1.8E-11	3.2E-12	9.4E-09	2.5E-07	3.8E-10	4.9E-08	3.9E-06	3.0E-08	3.2E-07
Am-241	1.1E-12	1.1E-12	7.9E-13	7.7E-13	1.8E-11	3.6E-07	6.0E-12	1.1E-12	3.1E-09	8.4E-08	1.3E-10	1.6E-08	1.3E-06	1.0E-08	1.1E-07
Cm-242	1.6E-09	1.6E-09	1.1E-09	1.1E-09	2.6E-08	4.7E-04	7.4E-09	1.5E-09	4.2E-06	5.5E-05	3.2E-08	2.1E-05	7.1E-04	1.9E-06	6.4E-06
Cm-243	8.9E-14	8.9E-14	6.4E-14	6.2E-14	1.5E-12	2.9E-08	4.8E-13	8.6E-14	2.5E-10	6.6E-09	9.8E-12	1.3E-09	1.0E-07	7.8E-10	7.8E-09
Cm-244	1.1E-11	1.1E-11	8.1E-12	7.9E-12	1.9E-10	3.7E-06	6.1E-11	1.1E-11	3.2E-08	8.2E-07	1.2E-09	1.7E-07	1.2E-05	9.6E-08	9.5E-07
I-131	3.9E+01	5.6E+01	3.1E+01	5.0E+01	7.0E+02	9.4E+06	5.2E+01	6.1E+02	0.0E+00	1.2E+06	2.8E+03	3.1E+06	1.6E-02	1.8E+06	1.9E+05
I-132	2.2E+02	3.1E+02	1.8E+02	2.8E+02	4.0E+03	3.5E+06	6.7E+01	3.4E+03	0.0E+00	1.3E+05	3.3E+01	5.9E+04	4.2E-05	7.3E+02	3.4E+03
I-133	1.2E+02	1.8E+02	9.9E+01	1.6E+02	2.2E+03	1.5E+06	1.8E+01	1.9E+03	0.0E+00	3.3E+05	1.8E+02	5.1E+05	7.9E-04	2.9E+05	1.2E+04
I-135	1.7E+02	2.5E+02	1.4E+02	2.2E+02	3.1E+03	4.0E+05	7.8E+00	2.7E+03	0.0E+00	1.7E+05	6.6E+01	1.3E+05	2.0E-04	7.5E+04	4.4E+03
H-3	6.3E+10	9.0E+10	5.0E+10	8.0E+10	1.1E+11	0.0E+00	0.0E+00	1.9E+09	0.0E+00	9.7E+09	1.6E+09	8.8E+08	2.0E+09	5.3E+09	0.0E+00

The discharge route is not continuous discharge but on batch discharge.

*: (1)Power Operation, (2)Start-up, (3)Shutdown, (4)Outage

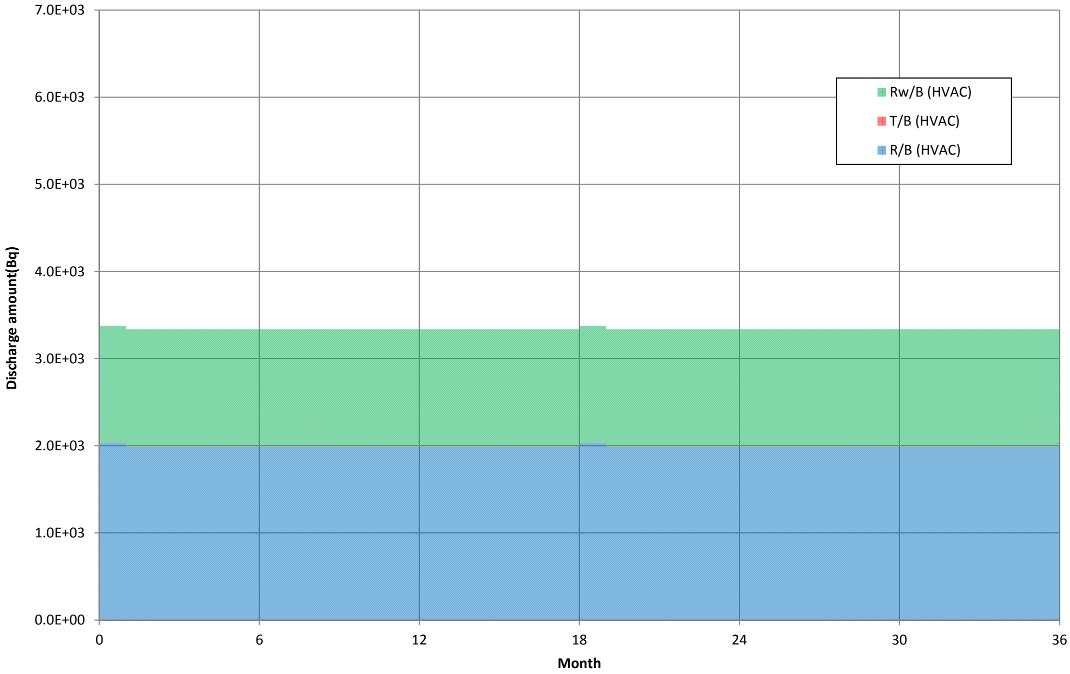

Figure 7.1-7: Predicted Monthly Gaseous Discharge of Particulate from the HVAC System

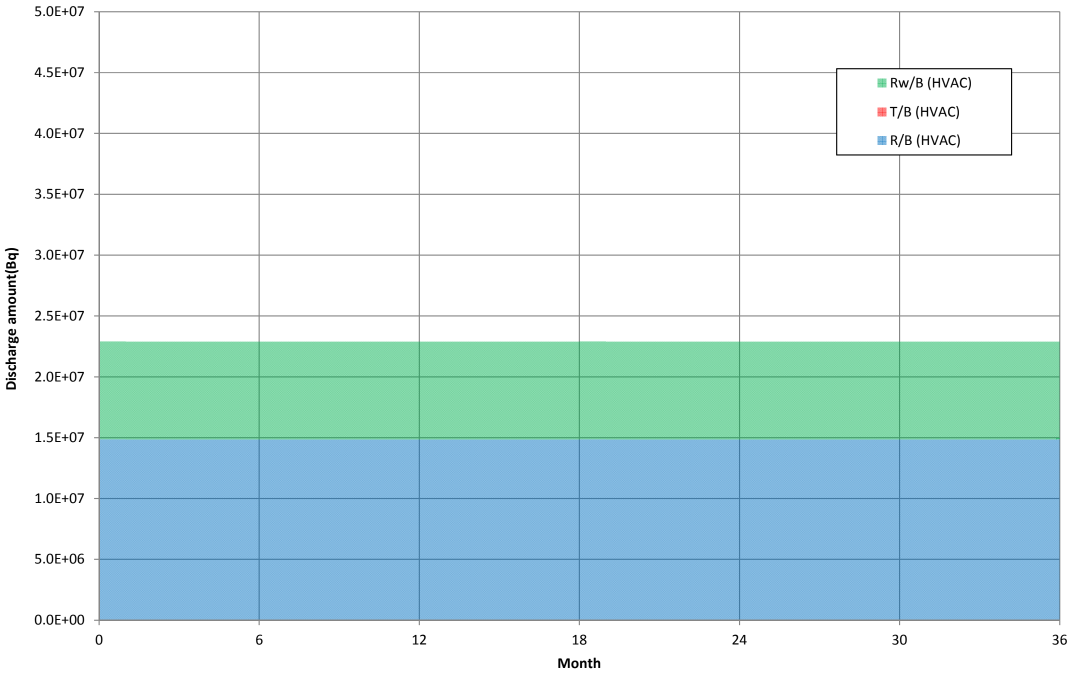

Figure 7.1-8: Predicted Monthly Gaseous Discharge of Iodine from the HVAC System

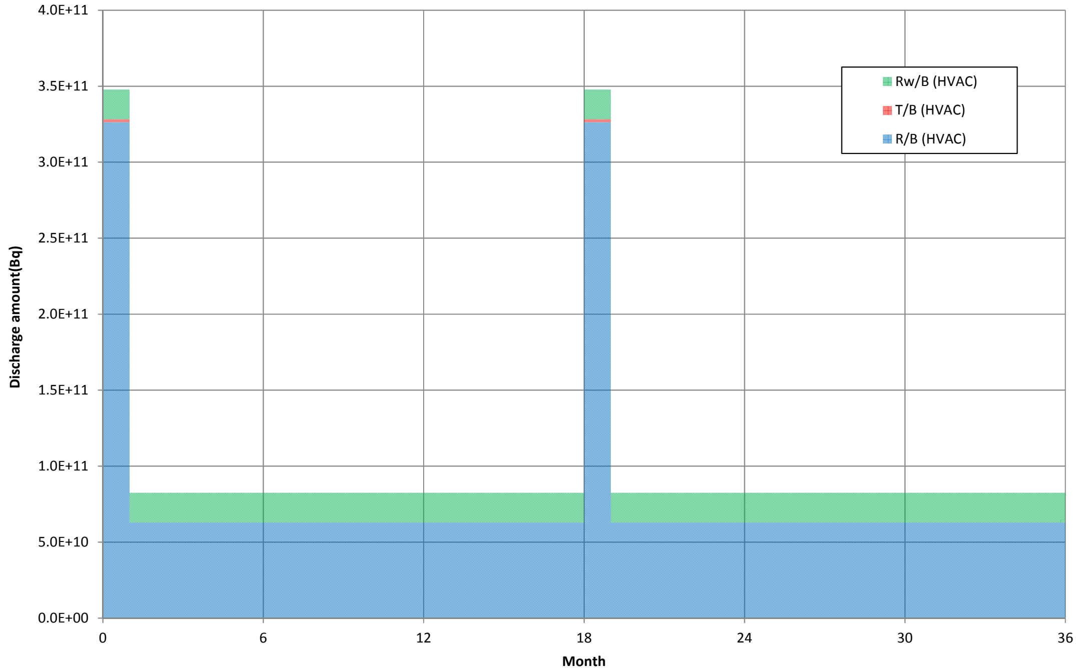

Figure 7.1-9: Predicted Monthly Gaseous Discharge of H-3 from the HVAC System

7.1.3. Monthly Gaseous Discharges – TGS System

A schematic diagram showing the discharge routes for gaseous discharge via the TGS system is shown in Figure 7.1-10.

During reactor operation such as start-up, power operation and shutdown, the discharge routes for gaseous discharge via TGS system are described in STEPs 1 to 2 below (the 'Item' numbers correspond with the numbers in Figure 7.1-10).

STEP1: For sealing the turbine shaft seal parts and the major valve gland parts by using steam, the water in Condensate Storage Tank (CST) is evaporated by gland steam evaporator, and the steam including tritium, iodine and other particulate nuclides migrate. *Item 1*

STEP2: The sealing steam including the radioactivity is condensed by gland steam condenser after sealing the turbine shaft seal parts and the major valve gland parts, and is discharged to the stack. *Items 2 and 3*

During outage, the TGS system is not operated because the operation of the turbine is subject to reactor operation. Therefore no discharge is made through the TGS system during outage.

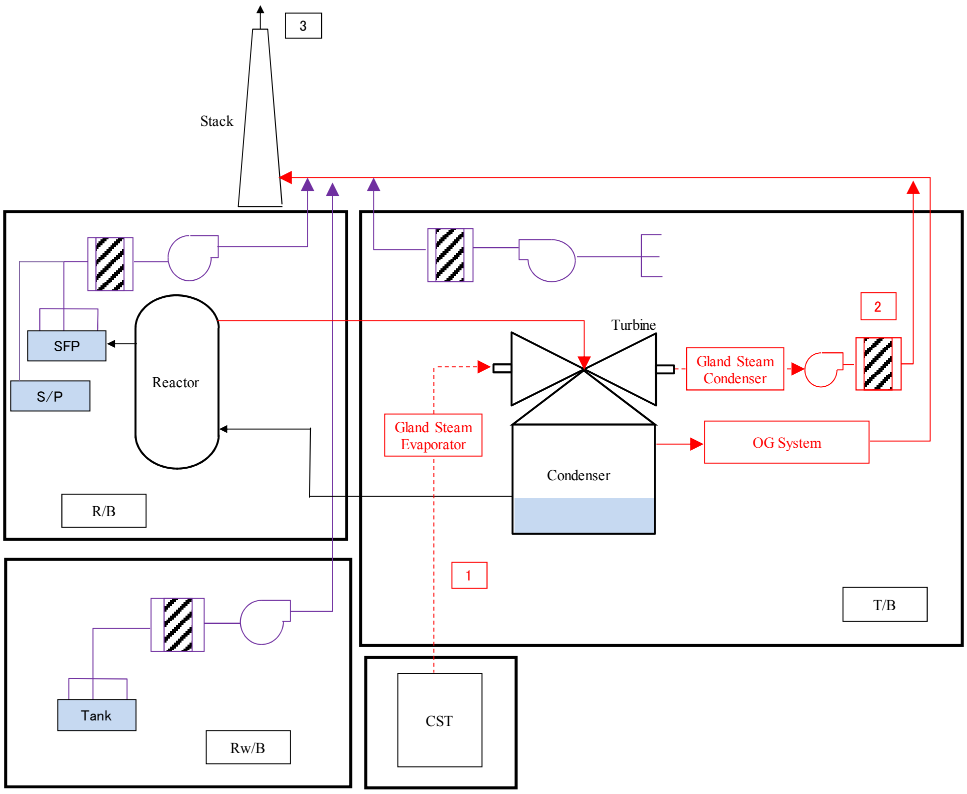

Figure 7.1-10: Schematic of Gaseous Discharges for the TGS system

The radioactive discharges from the TGS system are estimated by considering the evaporation of water from the CST. Tritium, iodine and other particulate nuclides migrate into the vapour phase when water from the CST is evaporated. The majority of the steam is condensed and particulates are removed by a High Efficiency Particulate Air (HEPA) filter, and the remaining gases are discharged from the TGS system via the stack.

Volatile non-condensable gases are removed from the reactor water/steam circuit at the SJAE and are discharged via the OG system as outlined in section 7.1.1. As a result, noble gases and C-14 are typically not present in the CST and so very low concentrations would be expected in the TGS system discharge. Therefore, for the purposes of this assessment the discharge of noble gases and C-14 from the TGS system is assumed to be zero.

The monthly discharges of radionuclides from the TGS system have been calculated based on the transport of radioactivity through the CST and the TGS system, shown in Figure 7.1-10 and Figure B8 of Appendix B. The calculations take account of the flow rate of radioactivity through these systems, including partitioning between water and air, water temperature, evaporation rates and removal by abatement systems.

The concentration of radionuclides in the CST is based on PrST calculation point CST-1-PO-BE and shown in Table 7.1-6 (a diagram showing the location of the calculation point is presented in Appendix B). The resulting monthly radioactive discharges from the TGS system have been determined using the data presented in [Ref-10] and are presented in Table 7.1-7.

Table 7.1-6: Radioactive Concentration in the CST

Radionuclides	Concentration in CST (Bq/cm^3) PrST: CST-1-PO-BE
Cr-51	5.2E-01
Mn-54	3.7E-01
Co-58	6.5E-01
Fe-59	1.0E-01
Co-60	6.2E-01
Zn-65	1.6E-01
Sr-89	2.1E-01
Sr-90	1.4E-02
Zr-95	2.8E-01
Nb-95	6.1E-01
Ag-110m	1.7E-04
Sb-122	2.6E-03
Sb-124	2.6E-01
Sb-125	5.3E-02
Cs-134	5.0E-02
Cs-137	3.0E-02
Ba-140	1.9E-01
La-140	2.2E-01
Ce-141	2.6E-01
Ce-144	2.4E-01
Pu-238	4.4E-08
Pu-239	5.6E-09
Pu-240	9.1E-09
Am-241	3.0E-09
Cm-242	2.5E-06
Cm-243	2.4E-10
Cm-244	3.0E-08
I-131	1.0E-01
I-132	4.0E-01
I-133	2.4E-01
I-135	3.6E-01
H-3	3.5E+02

Table 7.1-7: Monthly Discharge from the TGS System (1/2)

Radionuclides	Monthly Discharges during Power Operation (Bq/month)	Monthly Discharges during Outage (Bq/month)
Cr-51	1.9E+03	0.0E+00
Mn-54	1.4E+03	0.0E+00
Co-58	2.4E+03	0.0E+00
Fe-59	3.8E+02	0.0E+00
Co-60	2.3E+03	0.0E+00
Zn-65	5.8E+02	0.0E+00
Sr-89	7.8E+02	0.0E+00
Sr-90	5.0E+01	0.0E+00
Zr-95	1.0E+03	0.0E+00

Table 7.1-7: Monthly Discharge from the TGS System (2/2)

Radionuclides	Monthly Discharges during Power Operation (Bq/month)	Monthly Discharges during Outage (Bq/month)
Nb-95	2.2E+03	0.0E+00
Ag-110m	6.1E-01	0.0E+00
Sb-122	9.5E+00	0.0E+00
Sb-124	9.4E+02	0.0E+00
Sb-125	2.0E+02	0.0E+00
Cs-134	1.8E+02	0.0E+00
Cs-137	1.1E+02	0.0E+00
Ba-140	6.9E+02	0.0E+00
La-140	8.0E+02	0.0E+00
Ce-141	9.4E+02	0.0E+00
Ce-144	8.8E+02	0.0E+00
Pu-238	1.6E-04	0.0E+00
Pu-239	2.1E-05	0.0E+00
Pu-240	3.3E-05	0.0E+00
Am-241	1.1E-05	0.0E+00
Cm-242	9.0E-03	0.0E+00
Cm-243	8.7E-07	0.0E+00
Cm-244	1.1E-04	0.0E+00
I-131	3.7E+05	0.0E+00
I-132	1.5E+06	0.0E+00
I-133	8.9E+05	0.0E+00
I-135	1.3E+06	0.0E+00
H-3	1.3E+11	0.0E+00

N.B. No discharges of noble gases and carbon-14 are expected from the TGS system [Ref-10].

The monthly discharges of particulate, iodine and H-3 from the TGS system are depicted in Figures 7.1-11, 7.1-12 and 7.1-13 respectively (see comments under Figure 7.1-1 regarding the "idealised" nature of these graphs).

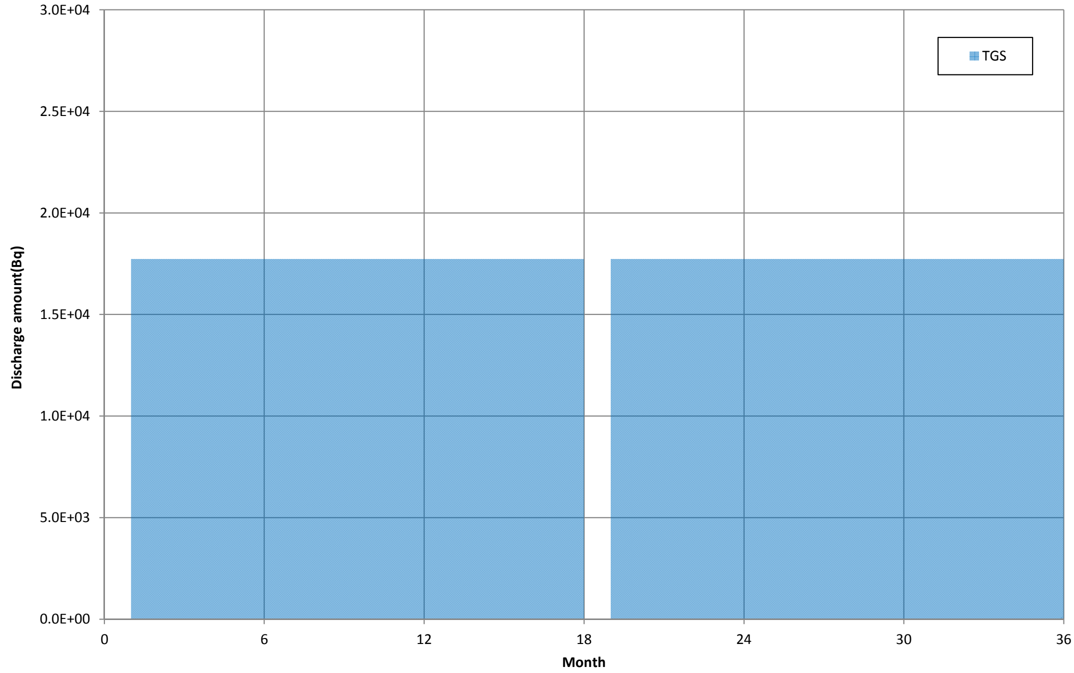

Figure 7.1-11: Predicted Monthly Gaseous Discharge of Particulate from the TGS System

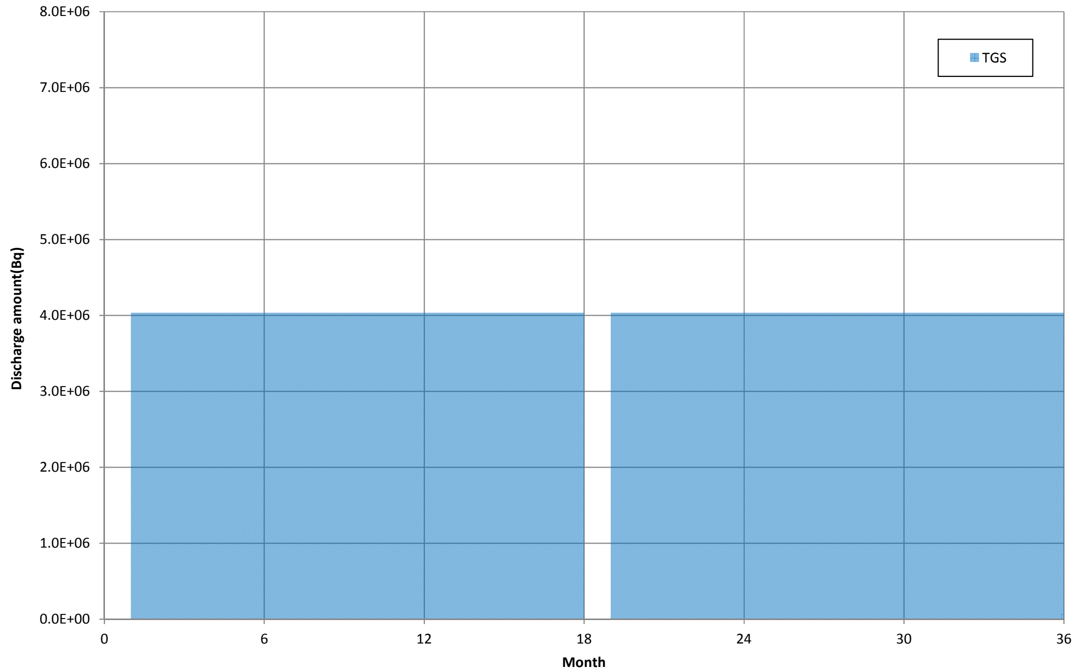

Figure 7.1-12: Predicted Monthly Gaseous Discharge of Iodine from the TGS System

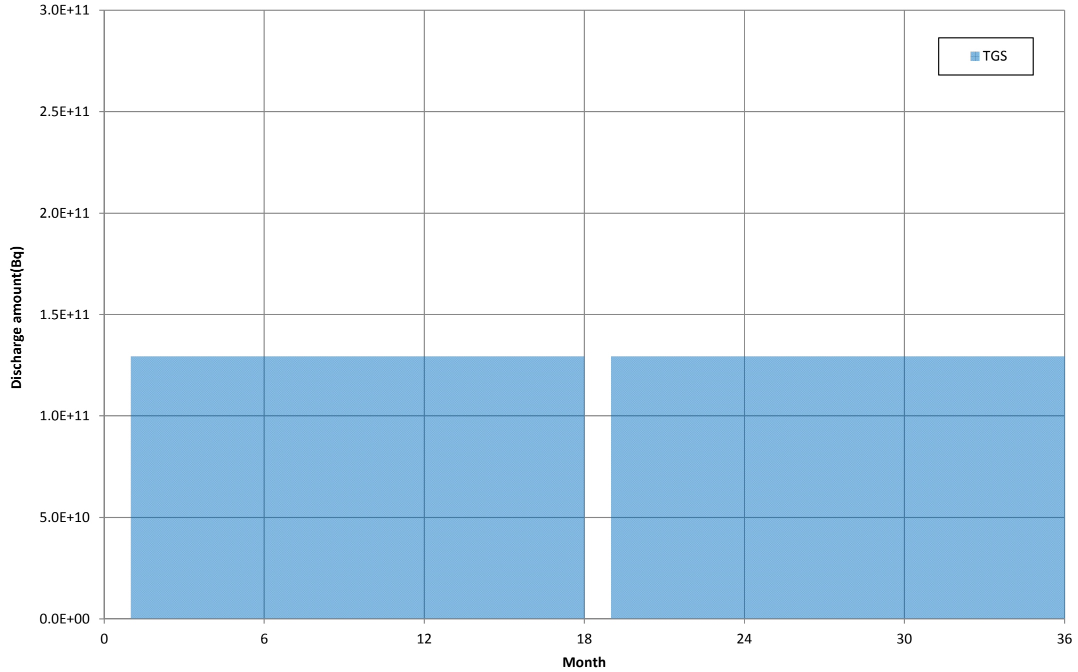

Figure 7.1-13: Predicted Monthly Gaseous Discharge of Tritium from the TGS System

7.1.4. Annual Gaseous Discharges - Rolling Twelve Monthly Basis

Based on the analysis presented above, Figures 7.1-14, 7.1-15, 7.1-16 and 7.1-17 show the gaseous discharges on a 12 month rolling basis, based on an 18 month operating cycle (17 months power operations and 1 month outage), i.e. an outage occurs at months 18, 36 and 54 in the following figures. Outages are indicated by the bold vertical lines on each figure.

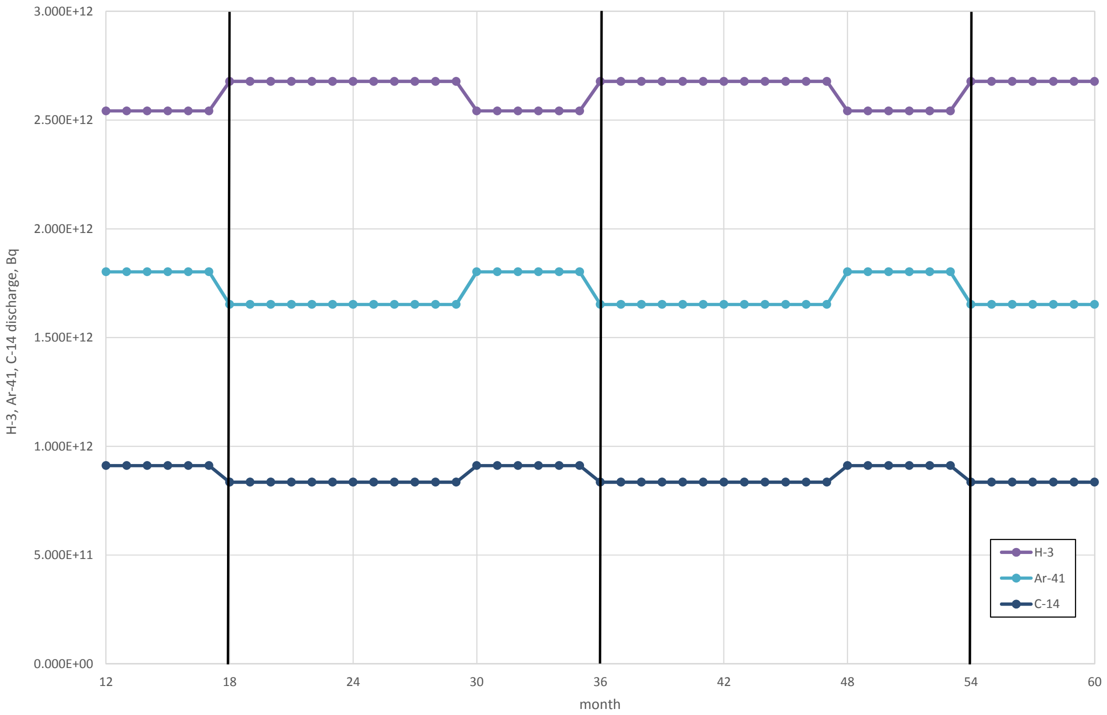

Figure 7.1-14: Rolling 12 Monthly Gaseous Discharges of Ar-41, H-3 and C-14

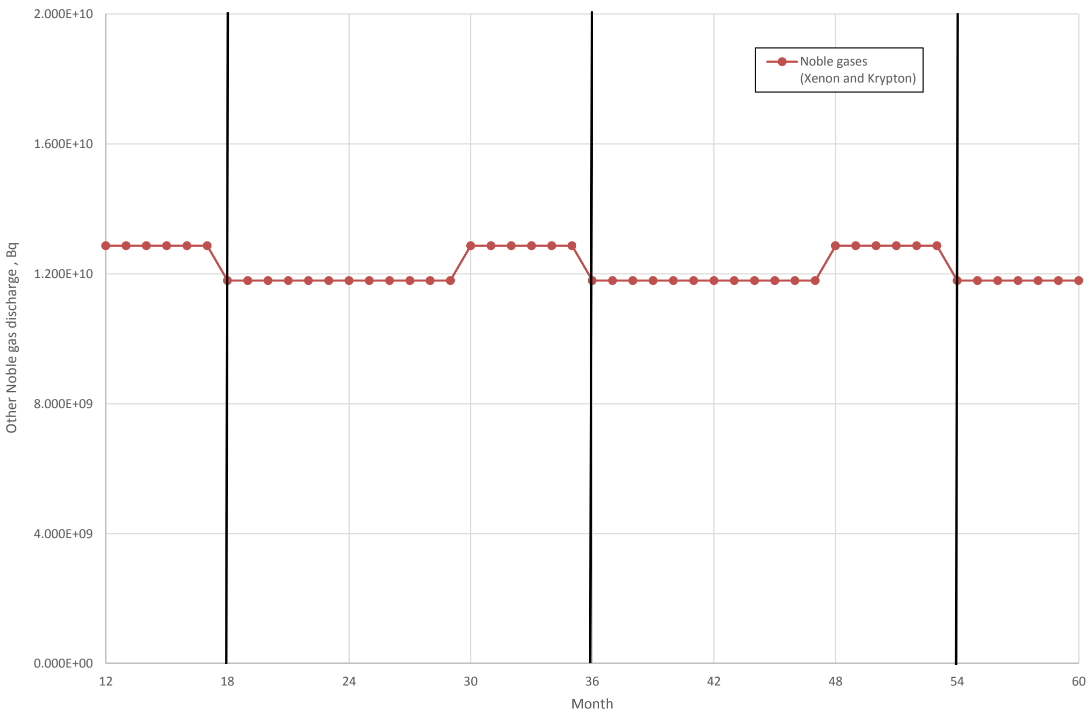

Figure 7.1-15: Rolling 12 Monthly Gaseous Discharge of Noble Gases (Xenon and Krypton)

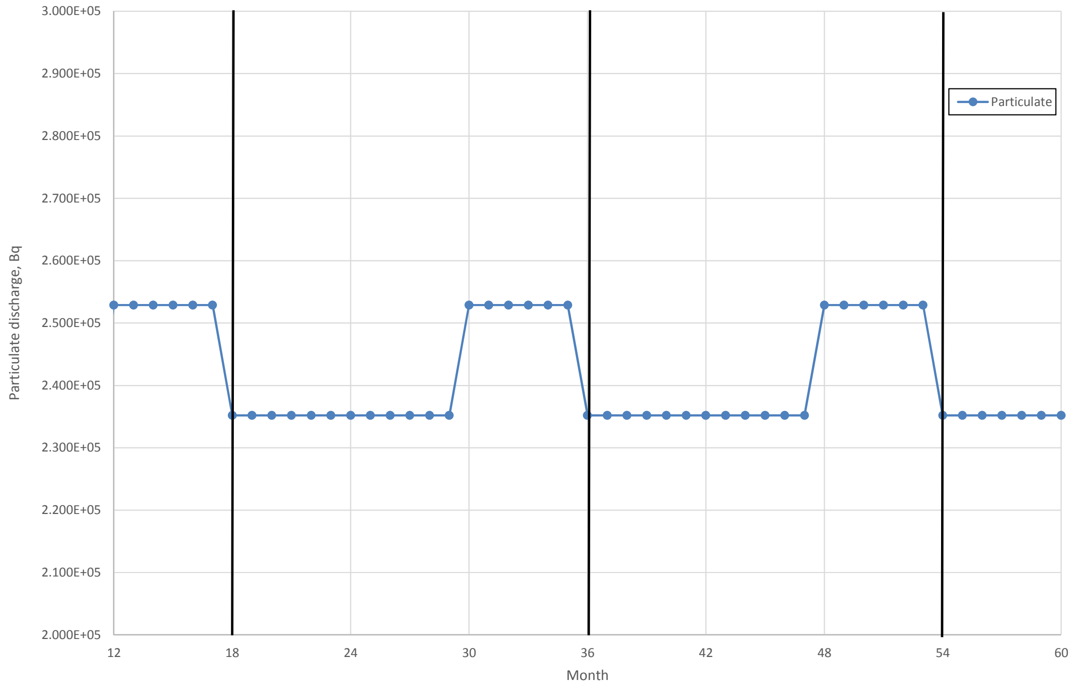

Figure 7.1-16: Rolling 12 Monthly Gaseous Discharge of Particulate

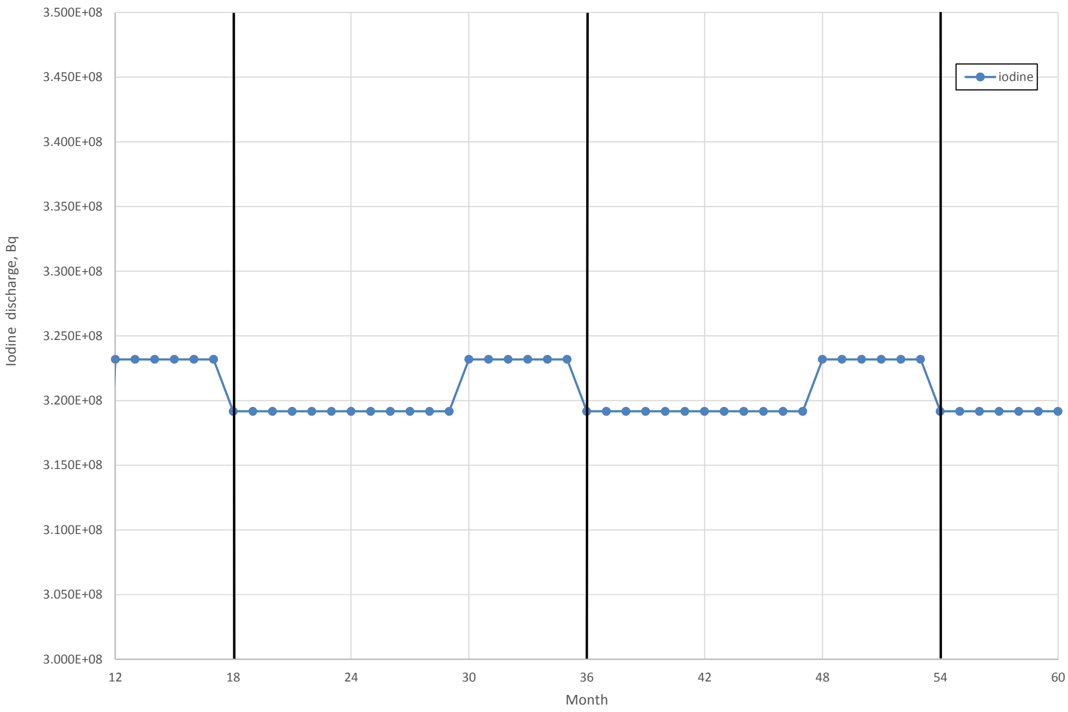

Figure 7.1-17: Rolling 12 Monthly Gaseous Discharge of Iodine

Figures 7.1-14 to 7.1-17 show the effect of the outage mode has on the rolling 12 monthly discharges for a 17 month operational + one month outage cycle.

For Ar-41, C-14 and noble gases, the discharges are made from the OG system. Therefore no discharges occur during the outage which means that the rolling 12 monthly discharge reduces by 1/12th for the 12 months that follow the outage. Once that 12 month period has elapsed, discharges occur in all the preceding months so the rolling annual discharge is at maximum value for 6 months until the next outage.

For H-3, particulates and iodine there are two contributions from HVAC system and TGS system. The discharge via HVAC system increases during the outage as additional systems are brought into use or are opened for maintenance, e.g. Reactor Well, D/S pool, Suppression Pool, Turbine Condenser well. The following rolling 12 monthly discharges include the higher monthly discharge for that one month outage. 12 months after the outage period the rolling 12 monthly discharge falls to the lower level until the next outage. However, during outages there is no discharge via TGS system and no discharge from the OG system. As the total contributions, the rolling annual discharges for tritium are at maximum value for 12 months including outage and for particulate and iodine are at maximum value for 6 months leading up to the next outage.

7.1.5. Annual Gaseous Discharges – Total

Based on the assumptions of maximum rolling 12 month discharges described in the previous section, the maximum annual gaseous discharges are shown in Table 7.1-8.

Table 7.1-8: Annual Gaseous Discharge Amounts from the Stack (1/2)

Radionuclides	Total(Bq/y)
Ar-41	1.8E+12
Kr-85	1.0E+08
Kr-85m	2.3E+09
Kr-87	2.3E+03
Kr-88	1.8E+08
Kr-89	0.0E+00
Xe-131m	1.4E+08
Xe-133	1.0E+10
Xe-133m	1.7E+06
Xe-135	1.7E-11
Xe-135m	0.0E+00
Xe-137	0.0E+00
Xe-138	0.0E+00
C-14	9.1E+11
Cr-51	3.1E+04
Mn-54	2.2E+04
Co-58	3.6E+04
Fe-59	5.8E+03
Co-60	3.7E+04
Zn-65	1.0E+04
Sr-89	9.9E+03

Table 7.1-8: Annual Gaseous Discharge Amounts from the Stack (2/2)

Radionuclides	Total(Bq/y)
Sr-90	6.3E+02
Zr-95	1.3E+04
Nb-95	2.8E+04
Ag-110m	9.5E+00
Sb-122	1.2E+02
Sb-124	1.2E+04
Sb-125	2.4E+03
Cs-134	2.3E+03
Cs-137	1.4E+03
Ba-140	8.6E+03
La-140	9.9E+03
Ce-141	1.2E+04
Ce-144	1.1E+04
Pu-238	2.3E-03
Pu-239	2.9E-04
Pu-240	4.6E-04
Am-241	1.6E-04
Cm-242	1.2E-01
Cm-243	1.2E-05
Cm-244	1.5E-03
I-131	1.9E+08
I-132	6.2E+07
I-133	4.3E+07
I-135	2.5E+07
H-3	2.7E+12

It is recognised some of the values in the above table are very small; for the purposes of setting the proposed limit the discharge of Xe-135 has been set to zero. (See the table 9.2-1)

7.1.6. Gaseous Discharges - Expected Event

In addition to the annual discharges derived from the month by month calculations for normal operations described above, the proposed annual limits must account for any additional activity released from any 'events that are expected to occur' (as defined in the Environment Agency's GDA P&ID [Ref-1]), i.e. expected events.

Hitachi-GE has undertaken a study to identify and assess the unplanned events that could have a bearing on discharges and that are considered likely to occur during normal operation of a UK ABWR across its planned plant lifetime [Ref-11]. The study identified one such expected event, specifically a fuel pin failure.

Should a fuel pin failure occur, elevated radioactivity, as a result of the escape of fission products from the fuel pin, is detected at the inlet of the OG system and a process is initiated to identify and isolate the fuel assembly in which the pin failure has occurred. Once the relevant fuel assembly is isolated, the fission within the fuel and subsequent leakage into the coolant reduces. This identification and isolation process can last up to 14 days [Ref-11].

Failure of a fuel pin increases the noble gas release ratio ('f-value'). Should a fuel pin failure occur, OPEX shows that a noble gas release ratio of 1.0E+08 Bq/s (the LCO value) is appropriate to calculate the associated discharges. The gaseous discharge due to the expected event was quantified by taking account of the escape of volatile fission products into the reactor water and their transport around the steam circuit, separation from the steam at the SJAE and subsequent retention in the OG system [Ref-11].

The concentration of radionuclides in the discharged gas is based on PrST calculation point OG-7-PO-LCO and shown in Table 7.1-9 (a diagram showing the location of the calculation point is presented in Appendix B).

Table 7.1-9: Radioactive Concentration in the OG System Exhaust for a Fuel Pin Failure Expected Event

Radionuclides	Concentration in Reactor Steam (Bq/cm^3) PrST: OG-7-PO-LCO
Kr-85	8.3E-02
Kr-85m	4.1E-01
Kr-87	3.7E-07
Kr-88	4.1E-02
Kr-89	0.0E+00
Xe-131m	1.9E-01
Xe-133	1.3E+01
Xe-133m	1.0E-03
Xe-135	2.3E-21
Xe-135m	0.0E+00
Xe-137	0.0E+00
Xe-138	0.0E+00

The amount of radioactivity calculated to be discharged from the Main Stack during an expected event [Ref-11] is presented in Table 7.1-10 below.

Table 7.1-10: Gaseous Discharges for a Fuel Pin Failure Expected Event

Radionuclides	Activity Discharge (Bq per event)
Kr-85	1.1E+09
Kr-85m	5.5E+09
Kr-87	5.0E+03
Kr-88	5.5E+08
Kr-89	0.0E+00
Xe-131m	2.6E+09
Xe-133	1.8E+11
Xe-133m	1.4E+07
Xe-135	3.1E-11
Xe-135m	0.0E+00
Xe-137	0.0E+00
Xe-138	0.0E+00

It is recognised some of the values in the above table are very small, and for the purposes of setting the proposed limit the discharge of Xe-135 has been set to zero. (See the Table 9.2-1)

As described in Section 6, liquid radioactive discharges are batched together and monitored in the final holding tank prior to discharge. Discharge of liquid waste is only permitted when monitoring data demonstrates that the discharge is at or below permitted values. Therefore even if the activity in liquid system due to expected events increases, the effect will result in a negligibly small increase in liquid discharges; and so this is not considered further.

7.2. Liquid Discharges

Monthly discharges and annual discharges are calculated in line with the requirements of the P&ID. The two main contributing radioactive systems are the HCW system and the laundry drain (LD) system as depicted in Figure B6 of Appendix B.

The HCW system is used to treat waste water from the hot lab drain in service building and the bottom drain from condensate demineraliser in T/B. The HCW system is designed to efficiently treat water with higher levels of impurities (both of soluble and insoluble) by passing it through an ion exchange bed and the evaporation process. After it is confirmed that residual activity levels are adequately low in HCW sample tank, the treated water is transferred usually to the CST for reuse within the plant as much as possible, and is discharged to the environment through a permitted route when the discharge is unavoidable due to the requirement to maintain the water balance in the power plant.

The LD system is used to treat waste water from the laundry and the personnel showers and hand washing facilities. This water contains detergents and organic impurities as well as crud with low levels of radiological contamination. Efficient removal of the detergents and organic material (as well as the radioactive crud) requires a different treatment process (with both filtration and activated carbon adsorption steps). After it is confirmed that residual activity levels are adequately low in LD sample tank, the treated water is discharged to the environment through a permitted route.

Discharges are also made from the controlled area drain (CAD) system. The CAD system is used to collect waste water from other plant and systems in the Radiologically Controlled Areas (RCAs) in the R/B and T/B. The waste water collected from the RCAs is potentially contaminated. This water is simply sampled to confirm it contains no significant radiological contamination (or unacceptable chemical contamination) before it is discharged to the environment. If the water is found to contain any significant radiological contamination or unacceptable chemical contamination, then the operator can route the water for treatment using the HCW [Ref-29]. However, it is during accident only that the waste water collected from the RCAs is contaminated because the waste water collected in the CAD system during normal operation consists of condensate water of local cooling units and coolant water blow from the R/B and T/B. Therefore, this discharge route is not considered in discharge assessment.

7.2.1. Monthly Liquid Discharges

Radioactive liquid discharges to the environment are made from the HCW sample tanks and the LD sample tanks. The liquid discharges from the contributing systems will vary depending on the operating mode, however, it is considered that the activity concentration in these tanks does not vary significantly over an operating cycle due to both the capability to recirculate the contents through the treatment systems (to ensure the discharge limits are met) and also the volume of the tanks themselves providing a buffering effect to any minor variations in the concentration of the feed water. Table 7.2-1 shows the relevant discharge route together with the type of ST used to calculate the discharge value.

Table 7.2-1: Summary of Contributing Discharge Routes, Operating Modes and Source Terms used to Calculate Liquid Discharges

N.B. The Operating Mode definitions, as used in Table 7.2-2 are: (1) - Power operation, (2) – Start-up, (3) – Shutdown, (4) – Outage.

Discharge Point to the Environment	Discharge Building	Discharge Route	PrST calculation point	Operating modes considered and type of ST used	Remarks
Canal* to the sea	Rw/B	HCW system	HCW sample tank	(1)(2)(3)(4): CA(BE)	(1)(2)(3)(4): The concentration of activity in the tanks does not change significantly over operational cycle due to the recirculation capability and the buffering effects of the tank, therefore, CA(BE) is used.
	S/B	LD system	LD sample tank		

*: Canal is the condenser cooling water outfall.

Whilst the liquid discharges from the UK ABWR are made on a batch basis, the frequency of the batch discharges is greater than once per week and they have been considered to be equivalent to a continuous discharge for the purposes of the radiological discharge and dose modelling assessments, and have been determined on an annual basis only.

The concentrations of radioactive species were taken from the following PrST calculation points (the locations of these calculation points are presented in schematic diagrams in Appendix B) and shown in Table 7.2-2:

HCW sample tank	HCW-6-CA-BE
LD sample tank	LD-8-CA-BE

Table 7.2-2: HCW and LD Activity Concentrations

Radionuclides	HCW Sample Tank Concentration (Bq/m^3) PrST value : HCW-6-CA-BE	LD Sample Tank Concentration (Bq/m^3) PrST value: LD-8-CA-BE
H-3	3.5E+08	1.3E-02
Cr-51	1.5E+01	1.8E-01
Mn-54	1.6E+01	4.0E+01
Fe-55	5.8E+01	1.0E+03
Fe-59	1.6E+00	1.8E+00
Co-58	2.2E+01	3.4E+00
Co-60	7.0E+01	7.1E+01
Ni-63	3.5E+00	9.4E+01
Zn-65	1.4E+01	8.1E+00
Sr-89	3.8E+00	6.9E-03
Sr-90	1.8E+00	4.5E-02
Zr-95	5.7E+00	7.6E+00
Nb-95	1.3E+01	1.7E+01
Ru-103	2.2E+00	2.4E+00
Ru-106	8.3E-01	1.9E+00
Ag-110m	1.8E-03	1.7E-04
Sb-122	1.3E-02	1.0E-02
Te-123m	1.3E-03	6.3E-03
Sb-124	2.2E+00	5.1E+00
Sb-125	1.2E+00	8.5E+00
I-131	6.2E+01	7.0E-04
Cs-134	2.5E+00	7.5E-03
Cs-137	2.7E+00	4.1E-02
Ba-140	2.7E+00	2.6E-03
La-140	3.1E+00	3.0E-03
Ce-141	4.0E+00	3.9E+00
Ce-144	1.1E+01	2.3E+01
Pu-238	7.0E-06	3.9E-04
Pu-239	9.1E-07	6.2E-05
Pu-240	1.5E-06	9.9E-05
Am-241	4.9E-07	1.2E-05
Cm-242	1.2E-04	1.9E-04
Cm-243	3.7E-08	5.3E-07
Cm-244	4.6E-06	5.0E-05

The calculated discharges of radionuclides from the HCW and LD systems have taken appropriate account of the movement of radioactivity through the respective systems including flow rates and effectiveness of abatement systems. It was also assumed in the discharge assessment that all of the contaminated liquid wastes processed in the HCW system and the LD system are discharged to the environment. For the purposes of this assessment, the annual discharge volume is taken as the maximum conceivable volumes that could be discharged from the HCW and LD systems into the environment; these are shown in Table 7.2-3 below.

Table 7.2-3: Environmental Annual Discharge Volume

Discharge Point	Environmental Annual Discharge Volume (m^3/y)
Outlet of HCW system	560
Outlet of LD system	2240

7.2.2. Annual Liquid Discharges – Total

The calculated annual discharges from the HCW and LD systems are presented in Table 7.2-4 below.

Table 7.2-4: Annual Liquid Discharges from the HCW and LD Sample Tanks (1/2)

Radionuclides	Annual Discharge from HCW Sample Tank (Bq/y)	Annual Discharge from LD Sample Tank (Bq/y)	Total (Bq/y)
H-3	2.0E+11	3.0E+01	2.0E+11
Cr-51	8.6E+03	4.1E+02	9.0E+03
Mn-54	9.0E+03	8.9E+04	9.8E+04
Fe-55	3.3E+04	2.3E+06	2.3E+06
Fe-59	9.0E+02	4.1E+03	5.0E+03
Co-58	1.2E+04	7.7E+03	2.0E+04
Co-60	3.9E+04	1.6E+05	2.0E+05
Ni-63	1.9E+03	2.1E+05	2.1E+05
Zn-65	7.9E+03	1.8E+04	2.6E+04
Sr-89	2.1E+03	1.5E+01	2.2E+03
Sr-90	1.0E+03	1.0E+02	1.1E+03
Zr-95	3.2E+03	1.7E+04	2.0E+04
Nb-95	7.1E+03	3.8E+04	4.5E+04
Ru-103	1.2E+03	5.3E+03	6.6E+03
Ru-106	4.7E+02	4.2E+03	4.7E+03
Ag-110m	1.0E+00	3.7E-01	1.4E+00
Sb-122	7.4E+00	2.3E+01	3.0E+01
Te-123m	7.3E-01	1.4E+01	1.5E+01
Sb-124	1.2E+03	1.1E+04	1.3E+04
Sb-125	6.8E+02	1.9E+04	2.0E+04
I-131	3.5E+04	1.6E+00	3.5E+04
Cs-134	1.4E+03	1.7E+01	1.4E+03
Cs-137	1.5E+03	9.2E+01	1.6E+03

Table 7.2-4: Annual Liquid Discharges from the HCW and LD Sample Tanks (2/2)

Radionuclides	Annual Discharge from HCW Sample Tank (Bq/y)	Annual Discharge from LD Sample Tank (Bq/y)	Total (Bq/y)
Ba-140	1.5E+03	5.9E+00	1.5E+03
La-140	1.7E+03	6.8E+00	1.7E+03
Ce-141	2.2E+03	8.8E+03	1.1E+04
Ce-144	6.0E+03	5.2E+04	5.8E+04
Pu-238	3.9E-03	8.7E-01	8.7E-01
Pu-239	5.1E-04	1.4E-01	1.4E-01
Pu-240	8.2E-04	2.2E-01	2.2E-01
Am-241	2.7E-04	2.6E-02	2.7E-02
Cm-242	6.8E-02	4.3E-01	5.0E-01
Cm-243	2.1E-05	1.2E-03	1.2E-03
Cm-244	2.6E-03	1.1E-01	1.1E-01

7.2.3. Annual Liquid Discharges - Rolling Twelve Month Basis

For the reasons outlined in section 7.2.1 there is no significant variation in the rolling 12 monthly discharges. Therefore no figures showing the rolling 12 monthly discharges are presented.

8. Headroom Factor

The preceding sections present the assessment of the anticipated discharges that may arise during the operation of the UK ABWR; they are based upon detailed theoretical modelling and assessment, supported by OPEX. However, as with the start-up of any newly built reactor, a degree of uncertainty remains in the annual discharges that may arise in the early years of operation of the UK ABWR. The Environment Agency guidance on limit setting [Ref-22] allows for operational flexibility between the optimised discharge level and the permitted discharge "limit" level. This allowance has been termed a "headroom factor". This section describes the approach to the determination of an appropriate headroom factor that will be applied to the discharge estimates in order to derive proposed annual discharge limits. It should be noted that the subsequent radiological dose modelling and assessment [Ref-12] will be based upon the discharge estimates multiplied by the headroom factor – i.e. the radiological dose assessment will be based upon the maximum permitted annual discharges.

Retrospective data for the UK ABWR is not available as it has not been built yet, therefore the original Environment Agency guidance on the approach to limit setting [Ref-23] which relies on use of directly comparative OPEX is not appropriate for use. As a result, an alternative approach has been adopted, and this is described in the following paragraphs.

To determine the headroom factor for the proposed UK ABWR annual discharge limits the mean and standard deviation for each of the datasets contributing to the calculation of the liquid and gaseous discharges [Ref-6] [Ref-7] are calculated. Once the mean and standard deviation are determined then a coverage factor *k*, can be applied to a normal distribution to determine the confidence interval of interest.

Typically one or two-sided distributions are considered depending on whether:

- an understanding of the spread of data around the mean value is required;. this would be based on the two sided distribution, e.g. counting uncertainty where values are usually expressed as the mean +/- *k* times the standard deviation; or,
- the likelihood of exceeding an upper value, expressed as the mean + *k* times the standard deviation, is required in which case the one sided distribution is applied; this is illustrated in Figure 8-1.

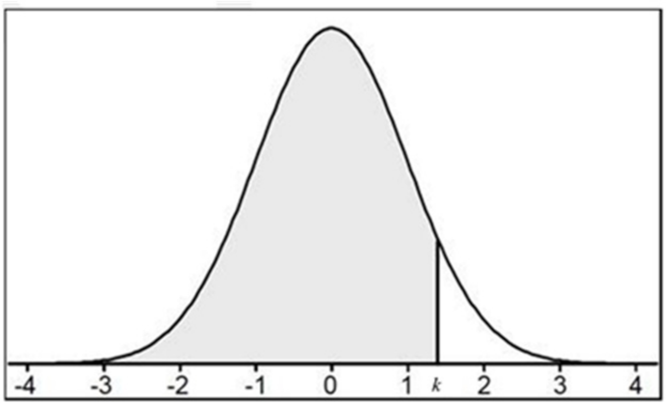

Figure 8-1: Illustration of the Cumulative Normal Distribution

It is considered appropriate that the coverage factor is based on a one sided normal distribution as an upper limit is being determined. Therefore to determine the headroom factor for the UK ABWR discharges, the following expression is used:

Headroom factor = (mean value + k × standard deviation) / mean value

Table 8-1 shows the coverage factor for a range of confidence intervals for a one-sided distribution.

Table 8-1: Confidence Interval and Coverage Factor, k

Confidence interval	Coverage factor, k(-)
90%	1.282
95%	1.645
97.5%	1.960
99%	2.326
99.5%	2.576
99.9%	3.090

8.1. Assumptions

i. Assumption of a normal distribution

Due to confidentiality and commercial reasons a limited amount of data is available on which the mean and standard deviation has been derived; the data sets are not sufficiently large to provide a definitive statement on the true distribution of the data. For this GDA assessment, a normal distribution has been selected over the potential alternative of a log-normal distribution, an illustration of which is shown in Figure 8.1-1.

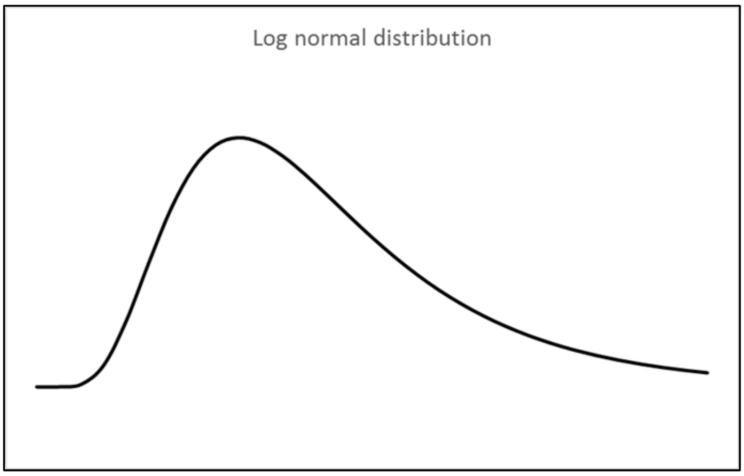

Figure 8.1-1: Illustration of the Log Normal Distribution

A number of fits to the available data have been examined including log normal distribution and polynomial functions. The only fits that were consistently better than normal distribution were higher order polynomials or sinusoidal functions which do not avail themselves to the methodology described in Section 8 for the derivation of the headroom factor. A normal distribution is a reasonable approach for the majority of the data sets. It is considered that a log-normal distribution would tend to bias the distribution toward lower values and have an extended tail for the higher values resulting in a lower value for the coverage factor k for the same one-sided confidence interval; it is considered that the use of the normal distribution would result in a more conservative estimate of the coverage factor, *k*. Therefore for this GDA assessment it is considered that, in view of the limited amount of OPEX data available, the use of the normal distribution is justifiable and results in a conservative estimate of the headroom factor.

ii. Representativeness of variation

The data on which the variation is based are taken from OPEX data [Ref-6][Ref-7]. OPEX data has been provided and expressed as either release rates into the coolant, activity concentrations in the reactor water, in the reactor steam or at the main stack. It is recognised that these absolute values in the UK ABWR systems do not necessarily represent the absolute values of the gaseous or liquid discharges, because no account of fractionation, carry over or the effect of clean up systems has been taken into account. However the variations in release rates, reactor water activity and reactor steam activity are expected to result in a linear relationship to the variations present in the processing and abatement systems and hence in gaseous and liquid discharges. This is because the behaviour of activity in the liquid/steam are expressed as a fraction of upstream vs downstream activity, e.g. carry over in steam into the turbines and into the OG system, the retention/hold up by abatement systems, whether they are charcoal beds, HEPA filters or ion exchange bed/filters. Therefore this will result in a linear relationship of the variation in reactor water/steam activity to the expected variability in the discharge data. Consequently, the relative variations in the activity of reactor water and steam are considered to be representative of the relative variations expected in the potential gaseous and liquid discharges, and it is considered they are appropriate to be used to determine the likely variation in gaseous and liquid discharges.

iii. Abatement efficiency

The design of the abatement plant that minimises the discharge of radioactive wastes from the UK ABWR to the environment, has been shown to represent BAT [Ref-15], and routine and preventative maintenance programmes, combined with routine performance monitoring, will ensure that the performance of the abatement plant will continue as designed throughout the UK ABWR's operating life. Any advances in abatement plant technology over the operating lifetime of the UK ABWR may result in upgrades to the installed abatement plant depending on whether it is BAT to do so. As a consequence, it is considered reasonable to assume that the performance of abatement systems remains constant throughout the operational life of the UK ABWR and the radioactive discharge and dose modelling assessments have not considered any medium to long term reduction in the effectiveness of the abatement plant.

iv. Duration of initial operations

At the site specific stage, it is expected that the liquid and gaseous discharges from an operational UK ABWR will be reviewed after several years and the use of actual discharge data may allow for a revision to the headroom factors described in this report (assuming they are adopted at site specific permitting), in order to reflect the actual variation in discharges achieved in the early years of operation.

8.2. Radionuclides Considered

Derivation of a headroom factor is not needed for all nuclides and only representative radionuclides or radionuclide groups need to be considered. Their selection is based upon their radiological significance, either in term of potential dose to members of the public, whether they result in a numerically large discharge to the environment or they are included in the list of radionuclide identified in EU2004 [Ref-19] as described in section 5.3. Based on this assessment, the radionuclides have been assigned to representative radionuclides or radionuclide groups for gaseous and liquid discharges and are listed in Table 8.2-1 below.

Table 8.2-1: Representative Radionuclides and Radionuclide Groups for Headroom Factor Derivation

Gaseous discharges	Liquid discharges
Noble gases excluding Ar-41 (based on Kr-88* OPEX)	Particulates (based on Co-60 OPEX)
Ar-41	Iodine (based on I-131 OPEX)
C-14	H-3
Particulates (based on Co-60* OPEX)	
Iodine (based on I-131* OPEX)	
H-3	

*: These radionuclides have been selected as representative nuclides of their respective groups from the perspective of dose impact.

8.3. Derivation of Headroom Factors

OPEX data presented in [Ref-6] and [Ref-7] were used to derive the average and standard deviation of the radionuclide groups of relevance; the results are shown in Table 8.3-1.

Table 8.3-1: Summary of the OPEX Data Average Values and Standard Deviations

Radionuclide group	Average	1 σ (-)
Noble gases excluding Ar-41 (based on Kr-88 OPEX)	9.9E+04 Bq/s	3.6E+04
Ar-41	8.4E+05 Bq/s	5.2E+05
C-14	2.9E+04 Bq/s	8.8E+03
Particulates (based on Co-60 OPEX)	1.0E+03 Bq/g	1.0E+03
Iodine (based on I-131 OPEX)	3.3E+03 Bq/s	7.1E+02
H-3	3.5E+02 Bq/g	3.2E+02

For noble gases (Kr-88), Ar-41, C-14 and iodine (I-131), the averages are based on the release rate (Bq/s) in main steam or the stack and for particulate (Co-60) and H-3, the averages are based on the concentration in reactor water.

A headroom factor corresponding to the 99th percentile means there would be a 1 in 100 chance that the permitted annual limit for a particular radionuclide or radionuclide group would be exceeded. Similarly, for the 99.9th percentile there is a 1 in 1000 chance that the annual limit would be exceeded. For the purposes of the GDA it is considered reasonable that the headroom factors for the representative radionuclide should be based on a coverage factor corresponding to the 99.9th percentile (see section 8.4); these are presented in Table 8.3-2.

Table 8.3-2: Headroom Factor based on the 99.9th percentile

Radionuclide group	Coverage factor, k (-)	Headroom factor (-)
Noble gases excluding Ar-41	3.09	2.1
Ar-41	3.09	2.9
C-14	3.09	1.9
Particulates	3.09	4.1
Iodine	3.09	1.7
H-3	3.09	3.8

8.4. Implications of Headroom Factor Adopted

Given a likely period of up to 5 years between commencement of operations of a UK ABWR and a formal review of the permitted annual discharges by the Environment Agency or NRW (depending on location and hence regulator), the implication of using headroom factors based on the 99.9th percentile of coverage is as follows:

- With a headroom factor corresponding to the 99.9th percentile there is a 1/1000 * 5 = 1 in 200 likelihood the permitted annual limit for a particular radionuclide or radionuclide group will be exceeded.

However it should be noted that there may be up to 5 independent permitted discharges(See section 9.4): potentially four in the case of gaseous discharges (noble gases, Ar-41, C-14, H-3) and one in the case of liquid discharges (H-3) then the likelihood of any exceedance of a permitted discharge over 5 year period is correspondingly altered as follows:

- With 5 independent permitted discharges, each using a headroom factor corresponding to the 99.9th percentile, there is a 1 in 40 likelihood that one of the permitted limits would be exceeded over a 5 year period.

8.5. Proposed Headroom Factor

It is proposed that the headroom factor adopted is based upon the 99.9th percentile value. Basing the headroom factor on the 99.9th percentile value results in a range of headroom factors between 1.7 to 4.1 (see Table 8.3-2) dependent upon the radionuclide group considered. The use of the 99.9th percentile value gives confidence that it is unlikely that a proposed annual limit would be exceeded in the early years of reactor operation, whilst allowing some flexibility between the optimised expected discharge and the permitted limit to the operator. At the site specific permitting stage, it is expected that the Environment Agency or NRW would review the discharge limits once suitable operating data has been acquired.

For comparison the “headroom factors”, i.e. the ratio between expected discharge and annual disposal limit, for the UK EPR [Ref-21] are 28 for noble gases, 2 for gaseous discharges of C-14, 17.5 for I-131 and 6 for gaseous discharges of H-3. For liquid discharges the “headroom factors” for the UK EPR are found to be 4 for C-14 and 1.4 for H-3.

9. Proposed Annual Discharge Limits

The proposed annual discharge limit for the UK ABWR is made up from two contributors: a contribution from the discharges made during the normal operation, and a second contribution from discharges made during the expected event.

A headroom factor is applied to the normal operation discharge, whereas no headroom factor is applied to the expected event discharge. No headroom factor is applied to the expected event discharge because:

- It has a relatively short duration.
- It is not as susceptible to variations in the operational state of the reactor.
- The quantification of its associated discharge is made using the LCO value [Ref-11] which means the value can be considered a limiting condition.

It is therefore appropriate to treat the expected event discharges as a discrete amount.

It is proposed that proposed annual limits for the UK ABWR are based on the following expression:

Proposed annual limit = (headroom factor × annual discharge) + expected event discharge.

The values used in the determination of the proposed annual limits are presented in Tables 9.2-1 and 9.2-2 for gaseous and liquid discharges, respectively.

9.1. Campaign Limits

The Environment Agency's P&ID [Ref-1] requires the requesting party to provide proposed annual limits for gaseous and liquid discharges and also give the requesting party the option to propose campaign limits; Hitachi-GE does not wish to propose any campaign limits for gaseous or liquid discharges.

9.2. Summary of Input Data

Table 9.2-1 and Table 9.2-2 present a summary of the data for each radionuclide present in the gaseous and liquid discharges respectively. Each table includes a brief description of the following:

- Radionuclide: the radionuclides listed are all those that may be released from an operation of a single unit.
- Production mechanism: a short description of the mechanism by which the radionuclide is produced, generating the activity e.g. Xe-133 is a fission product from fuel, structural Uranium.
- Route to the environment: the route by which each radionuclide is released to the environment is described.
- Estimated annual discharge: the total activity released during a rolling 12 month cycle which is expected maximum annual discharges in the UK ABWR.
- Headroom factor: this is the multiplication factor that Hitachi-GE proposes for the consideration of annual discharge limits for the UK ABWR.
- Proposed annual discharge limit: this is the limit that Hitachi-GE proposes for a single UK ABWR unit (Bq/y); the proposed annual discharge limit is calculated using the following expression: annual discharge × headroom factor + expected event discharge.
- Expected event discharges: this is the activity that will be released should an expected event occur (gaseous discharge only).
- Predicted dose at limits: the predicted maximum radiological dose likely to be received from operating a single UK ABWR at the proposed limits during a rolling 12 month cycle; the values in Table 9.2-1 and Table 9.2-2 have been calculated and reported in the Prospective Dose

Modelling report [Ref-12].

- Significant (Y/N): is the radionuclide considered 'Significant' (as per the methodology outlined in Section 9.3) and therefore selected for consideration as a radionuclide to which an annual discharge limit should be provided in the Environmental Permit.

Table 9.2-1: Summary Table for Gaseous Releases (1/2)

Radionuclide	Production mechanism	Route to the environment	Annual Discharge (Bq/y)	Headroom Factor	Expected Event Discharges (Bq)	Proposed Discharge Limit (Bq/y)	Dose at Limits (μSv/y)	Significant (Y/N)
Ar-41	Ar-40(n,g)Ar-41 Activation of entrained atmospheric Ar in coolant	Activation of coolant → Migration into steam → Separation at condenser → Discharge from the stack via OG system	1.8E+12	2.9	-	5.2E+12	7.5E-01	Y
Kr-85	Fission product from fuel, tramp Uranium	Migration into reactor water (direct or through pin failure) → 100 % migration into steam → Separation at condenser → Discharge via stack via OG system	1.0E+08	2.1	1.1E+09	1.3E+09	7.6E-07	Y
Kr-85m			2.3E+09		5.5E+09	1.0E+10	1.6E-04	
Kr-87			2.3E+03		5.0E+03	9.8E+03	9.3E-10	
Kr-88			1.8E+08		5.5E+08	9.3E+08	2.2E-04	
Kr-89			0.0E+00		0.0E+00	0.0E+00	0.0E+00	
Xe-131m			1.4E+08		2.6E+09	2.9E+09	2.4E-06	
Xe-133			1.0E+10		1.8E+11	2.0E+11	6.3E-04	
Xe-133m			1.7E+06		1.4E+07	1.8E+07	5.5E-08	
Xe-135			1.7E-11		3.1E-11	0.0E+00*	0.0E+00	
Xe-135m			0.0E+00		0.0E+00	0.0E+00	0.0E+00	
Xe-137			0.0E+00		0.0E+00	0.0E+00	0.0E+00	
Xe-138			0.0E+00		0.0E+00	0.0E+00	0.0E+00	
I-131	Fission product from fuel, tramp Uranium	Migration into reactor water (direct or through pin fracture) → Partial migration into steam→ Separation at condenser → Discharge via stack via OG system (negligible) Discharge of volatile Iodine in aqueous stream via HVAC system and TGS system	1.9E+08	1.7	-	3.2E+08	4.4E-01	N
I-132			6.2E+07		-	1.1E+08	7.5E-05	N
I-133			4.3E+07		-	7.3E+07	2.1E-03	N
I-135			2.5E+07		-	4.3E+07	8.1E-05	N
Sr-89	Fission product from fuel, trampUranium	Migration into reactor water (direct or through pin failure) → Entrainment of aerosol into steam lines→ Discharge via condenser and stack(negligible) Discharge of scattered particulate in aqueous stream via HVAC system Discharge of evaporated particulate in aqueous stream via TGS system	9.9E+03	4.1	-	4.1E+04	5.4E-07	N
Sr-90			6.3E+02		-	2.6E+03	6.4E-07	N
Zr-95			1.3E+04		-	5.3E+04	1.2E-06	N
Nb-95			2.8E+04		-	1.1E+05	7.4E-07	N
Cs-134			2.3E+03		-	9.4E+03	3.0E-06	N
Cs-137			1.4E+03		-	5.7E+03	2.4E-06	N
Ba-140			8.6E+03		-	3.5E+04	3.9E-07	N
La-140			9.9E+03		-	4.1E+04	8.9E-08	N
Ce-141			1.2E+04		-	4.9E+04	1.9E-07	N
Ce-144			1.1E+04		-	4.5E+04	2.1E-06	N

*：The proposed discharge limit is set to zero, as the annual discharge and expected event discharge for these radionuclide are nearly zero.

Table 9.2-1: Summary Table for Gaseous Releases (2/2)

Radionuclide	Production mechanism	Route to the environment	Annual Discharge (Bq/y)	Headroom Factor	Expected Event Discharges (Bq)	Proposed Discharge Limit (Bq/y)	Dose at Limits (µSv/y)	Significant (Y/N)
Pu-238	Activation of fuel U by neutron capture	Migration into reactor water (direct or through pin fracture) → Entrainment of aerosol into steam lines → Discharge via condenser and stack(negligible) Discharge of scattered particulate in aqueous stream via HVAC system Discharge of evaporated particulate in aqueous stream via TGS system	2.3E-03	4.1	-	9.4E-03	4.2E-10	N
Pu-239	Activation of fuel U by neutron capture		2.9E-04		-	1.2E-03	6.0E-11	N
Pu240	Activation of fuel U by neutron capture		4.6E-04		-	1.9E-03	9.4E-11	N
Am-241	Activation of fuel U by neutron capture		1.6E-04		-	6.6E-04	2.8E-11	N
Cm-242	Activation of fuel U by neutron capture		1.2E-01		-	4.9E-01	2.6E-09	N
Cm-243	Activation of fuel U by neutron capture		1.2E-05		-	4.9E-05	1.6E-12	N
Cm-244	Activation of fuel U by neutron capture		1.5E-03		-	6.2E-03	1.7E-10	N
H-3	Ternary fission in fuel B-10 (n,2a) H-3 from Boron in control rods (negligible) H-2 (n,g) H-3 from H-2 in reactor water	Migration into reactor water (direct or through pin failure or diffusion through pin cladding) → Entrainment of aerosol into steam lines → Discharge via condenser and stack(negligible) Evaporative losses from sources of tritiated water → Discharge via HVAC system and TGS system	2.7E+12	3.8	-	1.0E+13	1.2E+00	Y
C-14	N-14 (n,p) C-14 O-17 (n,a) C-14 both from fuel and reactor water C-13 (n,g) C-14 from structural materials	C-14 is always carried by stable carbon compounds. The air entrained in the coolant is ejected from the main condenser. This off-gas is fundamentally air, and therefore carbon, as carbon dioxide, exists in the similar ratio to other constituents as it does in air and discharged via OG system.	9.1E+11	1.9	-	1.7E+12	2.1E+01	Y
Cr-51	Cr-50 (n,g) Cr-51 Activation of reactor components Activation of insoluble and soluble metal crud and particulate in reactor water	Entrainment of aerosol into steam lines→ Discharge via condenser and stack (negligible) Discharge of scattered particulate in aqueous stream via HVAC system Discharge of evaporated particulate in aqueous stream via TGS system	3.1E+04	4.1	-	1.3E+05	3.1E-08	N
Mn-54	Fe-54 (n,p) Mn-54 Mn-55 (n,2a) Mn-54 Activation product		2.2E+04		-	9.0E+04	4.7E-06	N
Co-58	Ni-58 (n,p) Co-58 Activation product		3.6E+04		-	1.5E+05	2.3E-06	N
Fe-59	Fe-58 (n,g) Fe-59 Activation product		5.8E+03		-	2.4E+04	4.3E-07	N
Co-60	Co-59 (n,g) Co-60 Activation of reactor components Activation of insoluble and soluble metal crud and particulate in reactor water		3.7E+04		-	1.5E+05	7.8E-05	N
Zn-65	Zn-64 (n,g) Zn-65 Activation product		1.0E+04		-	4.1E+04	1.4E-05	N
Ag-110m	Ag-109 (n,g) Ag-110m Activation product		9.5E+00		-	3.9E+01	8.5E-09	N
Sb-122	Activation of Sb present in structural components such as bearings		1.2E+02		-	4.9E+02	2.1E-07	N
Sb-124			1.2E+04		-	4.9E+04	2.1E-05	N
Sb-125			2.4E+03		-	9.8E+03	6.4E-07	N
						total	2.4E+01	

Table 9.2-2: Summary Table for Liquid Releases (1/2)

Radionuclide	Production mechanism	Route to the environment	Annual Discharge (Bq/y)	Headroom Factor	Proposed Discharge Limit (Bq/y)	Dose at Limits (µSv/y)	Significant (Y/N)
Ru-103	Fission product from fuel, structural Uranium	Migration into reactor water (direct or through pin failure) → Partial migration into steam → Build-up in reactor, fuel pool water, etc → Liquid waste gathered in each sump → Discharge via LD sample tank- occasional Discharge via HCW sample tank- occasional	6.6E+03	4.1	2.7E+04	1.8E-08	N
Ru-106			4.7E+03		1.9E+04	7.0E-08	N
Te-123m			1.5E+01		6.2E+01	7.2E-10	N
Sr-89			2.2E+03		9.0E+03	1.0E-09	N
Sr-90			1.1E+03		4.5E+03	2.1E-09	N
Zr-95			2.0E+04		8.2E+04	5.5E-07	N
Nb-95			4.5E+04		1.8E+05	3.0E-07	N
I-131		Migration into reactor water (direct or through pin failure) → Partial migration into steam → Build-up in reactor and fuel pool water → Liquid waste gathered in each sump → Discharge via LD sample tank- occasional Discharge via HCW sample tank- occasional	3.5E+04	1.7	6.0E+04	1.2E-08	N
Cs-134	Fission product from fuel, structural Uranium	Migration into reactor water (direct or through pin failure) → Partial migration into steam → Build-up in reactor, fuel pool water, etc → Liquid waste gathered in each sump → Discharge via LD sample tank- occasional Discharge via HCW sample tank- occasional	1.4E+03	4.1	5.7E+03	5.3E-08	N
Cs-137			1.6E+03		6.6E+03	7.6E-08	N
Ba-140			1.5E+03		6.2E+03	2.4E-09	N
La-140			1.7E+03		7.0E+03	8.1E-10	N
Ce-141			1.1E+04		4.5E+04	5.9E-09	N
Ce-144			5.8E+04		2.4E+05	2.8E-07	N
Pu-238	Activation of fuel U by neutron capture	Migration into reactor water (direct or through pin failure) → Partial migration into steam → Build-up in reactor, fuel pool water, etc → Liquid waste gathered in each sump → Discharge via LD sample tank- occasional Discharge via HCW sample tank- occasional	8.7E-01	4.1	3.6E+00	4.4E-10	N
Pu-239			1.4E-01		5.7E-01	7.5E-11	N
Pu-240			2.2E-01		9.0E-01	1.2E-10	N
Am-241			2.7E-02		1.1E-01	6.0E-13	N
Cm-242			5.0E-01		2.1E+00	4.7E-13	N
Cm-243			1.2E-03		4.9E-03	1.1E-13	N
Cm-244			1.1E-01		4.5E-01	9.4E-13	N

Table 9.2-2: Summary Table for Liquid Releases (2/2)

Radionuclide	Production mechanism	Route to the environment	Annual Discharge (Bq/y)	Headroom Factor	Proposed Discharge Limit (Bq/y)	Dose at Limits (µSv/y)	Significant (Y/N)
H-3	Ternary fission in fuel B-10 (n,2a) H-3 (from Boron in control rods-negligible) H-2 (n,g) H-3 (from H-2 in reactor water)	Migration into reactor water (direct or through pin failure) → Partial migration into steam → Build-up in reactor, fuel pool water, etc → Liquid waste gathered in each sump → Discharge via LD sample tank- occasional Discharge via HCW sample tank- occasional	2.0E+11	3.8	7.6E+11	5.2E-05	Y
Cr-51	Cr-50 (n,g) Cr-51 Activation of reactor components, insoluble and soluble metal crud and particulate in reactor water	Migration into reactor water (direct or through pin failure) → Partial migration into steam → Build-up in reactor, fuel pool water, etc → Liquid waste gathered in each sump → Discharge via LD sample tank- occasional Discharge via HCW sample tank- occasional	9.0E+03	4.1	3.7E+04	1.7E-09	N
Mn-54	Fe-54 (n,p) Mn-54 Mn-55 (n,2a) Mn-54 Activation product		9.8E+04	4.1	4.0E+05	7.1E-06	N
Fe-55	Fe-54(n,g)Fe-55 Fe-56(n,2n)Fe-55		2.3E+06	4.1	9.4E+06	2.2E-07	N
Co-58	Ni-58 (n,p) Co-58 Activation product		2.0E+04	4.1	8.2E+04	4.4E-07	N
Fe-59	Fe-58 (n,g) Fe-59 Activation product		5.0E+03	4.1	2.1E+04	7.9E-08	N
Co-60	Co-59 (n,g) Co-60 Activation of reactor components, insoluble and soluble metal crud and particulate in reactor water		2.0E+05	4.1	8.2E+05	1.8E-04	N
Ni-63	Ni-62 (n,g) Ni-63		2.1E+05	4.1	8.6E+05	2.4E-07	N
Zn-65	Zn-64 (n,g) Zn-65 Activation product		2.6E+04	4.1	1.1E+05	2.9E-05	N
Ag-110m	Ag-109 (n,g) Ag-110m Activation product		1.4E+00	4.1	5.7E+00	1.8E-09	N
Sb-122	Activation of Sb present in structural components such as bearings		3.0E+01	4.1	1.2E+02	1.4E-09	N
Sb-124			1.3E+04	4.1	5.3E+04	6.1E-07	N
Sb-125			2.0E+04	4.1	8.2E+04	1.8E-07	N
					Total	2.7E-04	

9.3. Selection of Significant Radionuclides for Permitting

The basis upon which radionuclides are selected to be permitted for discharge by the Environment Agency is published in Environment Agency guidance [Ref-22]. The guidance states that the Environment Agency, or NRW,will normally set annual site limits for each radionuclide, or group of radionuclide(s) that meet certain parameters during normal operation, namely for radionuclides or groups of radionuclides that:

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 μSv per year);

Hitachi-GE has undertaken prospective radiological dose modelling to assess the impact to humans for all radionuclides that may be released at the proposed limits for the UK ABWR. The prospective radiological dose modelling was undertaken using the proposed limits for each radionuclide in order to provide a cautious assessment and include the widest possible range of radionuclides. In following this approach, and as shown in Table 9.2-2, there are no radionuclides that are in the liquid discharge that fall into this category. However, as presented in Table 9.2-1, H-3 and C-14 discharged via the gaseous discharge route are predicted to have indicative doses in excess of 1 μSv/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 μGy/h);

The threshold of 40 μGy/h presented above is for exposure from all sources of radiation. For a single source of exposure a more appropriate threshold is 10 μGy/h. There are no radionuclides that fall into this category, as demonstrated by the prospective dose modelling that Hitachi-GE has undertaken [Ref-12]. Hitachi-GE undertook the prospective radiological dose modelling to assess the impact to non-human species for all radionuclides that may be released from the UK ABWR, at the proposed limits. This was a cautious approach which included the widest possible range of radionuclides.

c. are significant in terms of the quantity of radioactivity discharged (that is, the discharge of a radionuclide exceeds 1 TBq per year);

Hitachi-GE has estimated discharges from a single UK ABWR unit using the approaches outlined in section 5, and has developed a suite of proposed limits for the single unit (as summarised in Table 9.2-1 and Table 9.2-2). As can be seen, the data shows that there are a series of individual radionuclides, if discharged at the proposed limits for 12 months, would exceed the 1TBq per year threshold for significance. These are:

- Gaseous H-3 with a predicted annual discharge of 10 TBq, if discharged at the proposed annual limit for 12 months;
- Gaseous C-14 with a predicted annual discharge of 1.7 TBq, if discharged at the proposed annual limit for 12 months;
- Gaseous Ar-41 with a predicted annual discharge of 5.2 TBq, if discharged at the proposed annual limit for 12 months; and
- Liquid H-3 with a predicted annual discharge of 0.76 TBq, if discharged at the proposed annual limit for 12 months. Although this discharge strictly is below the criterion for identification as a significant radionuclide it has been included because the gaseous H-3 discharge exceeds the

criterion.

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);

The International Atomic Energy Agency (IAEA) considers that practices giving rise to collective doses below 1 man Sievert may be exempted from regulatory control, only those radionuclides that exceed this value are to be included within this parameter. As demonstrated in the Prospective Dose Modelling report [Ref-12], the predicted collective dose from the single UK ABWR is within the 1 man Sievert threshold for liquid discharges for all population groups. For gaseous discharges the 1 man Sievert threshold is predicted to be exceeded for EU and World populations.

For these groups the radionuclide making up at least 95% of the collective dose is C-14. C-14 is already included on the list of significant radionuclides from parameters a) and c).

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

No radionuclides in the proposed liquid and gaseous discharges are constrained under national or international agreements or are of concern internationally. Therefore no additional radionuclides will be included within the selection of significant radionuclides as a result of this requirement.

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

Hitachi-GE considers that the following radionuclides are indicators of plant performance:

- For gaseous streams the noble gases concentration (enhanced noble gas activity would indicate possible fuel pin failure or failure of the OG system). In regard of liquid discharges, no additional radionuclides have been identified that would act as indicators of plant performance.

g. for the appropriate generic categories from the RSR Pollution Inventory (e.g. ‘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.

No additional radionuclides will be included within the selection of significant radionuclides as a result of this requirement.

9.4. UK ABWR Proposed Annual Limits

Tables 9.4-1 and 9.4-2 show the proposed annual limits for the UK ABWR gaseous and liquid discharges respectively. The limits have been derived by either direct use of the proposed limit for individual isotopes or summing the proposed limits for significant radionuclides as described in section 9.3, and are based on the annual discharges as described in section 7.1 for gaseous discharges, and section 7.2 for liquid discharges.

Other candidate radionuclides or radionuclide groups have not been proposed to be subject to limits, including gaseous discharges of iodine and gaseous and liquid discharges of other radionuclides. The reasons for not including these radionuclides or radionuclide groups are that they are not considered to meet the significance criteria defined in section 9.3. It is also worth noting that other radionuclides in the liquid waste streams are too low to be measured at point of discharge, or at locations upstream of the discharge point.

Table 9.4-1: Discharges to Air

Radionuclide (or group)	Derivation	Proposed Annual Limit – 12 month rolling (Bq)
Ar-41	As presented in table 9.2-1	5.2E+12
C-14	As presented in table 9.2-1	1.7E+12
H-3	As presented in table 9.2-1	1.0E+13
Noble gases, excluding Ar-41	Summation of Krypton and Xenon isotopes presented in table 9.2-1	2.2E+11

Table 9.4-2: Discharges to Liquid

Radionuclide (or group)	Derivation	Proposed Annual Limit – 12 month rolling (Bq)
H-3	As presented in table 9.2-2	7.6E+11

9.5. Comparison with International Plants

A review of annual discharges of radioactive materials from similar reactor types has been undertaken: data from operating European and US BWRs has been collated and a comparison has been made between the expected discharges from the UK ABWR and the reported annual discharges from operating BWRs, normalised to annual GWeh. The annual discharges from the European BWRs have been obtained from the European Commission RAdioactive Discharges Database (RADD) [Ref-25]. The RADD records the declared discharges which are based on the EU 2004 methodology [Ref-19] and therefore are considered to be based on measured data. The discharge data was extracted for the period 2005 to 2014. The annual reactor power, expressed as GWeh was extracted from the International Atomic Energy Agency's Power Reactor information System (PRIS) [Ref-26] for the same period. For the US BWRs and European reactor (Leibstadt) not included in the RADD then data presented in [Ref-24][Ref-27] was used.

The reactors considered in the review are shown in Table 9.5-1.

Table 9.5-1: List of BWR Selected for the Review

Reactor name	Reactor design	Commercial operation start	Reference Unit Power (GWe)	Country location
Europe BWRs				
Olkiltuoto	ABB-III, BWR-2500	Unit 1: 1979 Unit 2: 1982	0.88	Finland
Gundrennigen B+C	Type-72	Unit B: 1984 Unit C: 1985	Unit B: 1.284 Unit C: 1.288	Germany
Isar 1*	Type-69	1979	0.878	Germany
Philippsburg 1*	Type-69	1980	0.89	Germany
Cofrentes	BWR-6	1985	1.064	Spain
Santa Maria de Garona	BWR-3	1971	0.446	Spain
Forsmark	Unit 1: ABB-III, BWR-2500 Unit 2: ABB-III, BWR-2500 Unit 2: ABB-III, BWR-3000	Unit 1: 1980 Unit 2: 1981 Unit 3: 1985	Unit 1: 0.984 Unit 2: 1.12 Unit 3: 1.167	Sweden
Oskershamn	Unit 1: ABB-I Unit 2: ABB-II Unit 3: ABB-III, BWR-3000	Unit 1: 1972 Unit 2: 1975 Unit 3: 1985	Unit 1: 0.473 Unit 2: 0.638 Unit 3: 1.4	Sweden
Ringhals 1	ABB-I	1976	0.881	Sweden
Leibstadt	BWR-6	1984	1.22	Switzerland
US BWRs				
Clinton	BWR-6 (Mark 3)	1987	1.065	US
Grand Gulf-1	BWR-6 (Mark 3)	1985	1.419	US
LaSalle County-1	BWR-5 (Mark 2)	1984	1.137	US
LaSalle County-2	BWR-5 (Mark 2)	1984	1.14	US
Limerick-1	BWR-4 (Mark 2)	1986	1.13	US
Limerick-2	BWR-4 (Mark 2)	1990	1.134	US
Nine Mile Point-2	BWR-5 (Mark 2)	1988	1.276	US
Perry-1	BWR-6 (Mark 3)	1987	1.256	US
Susquehanna-1	BWR-4 (Mark 2)	1983	1.257	US
Susquehanna-2	BWR-4 (Mark 2)	1985	1.257	US

* power generation terminated in 2011

The BWRs listed in Table 9.5-1 were selected because the nuclear power plants are currently operating or have only recently ceased operating, their discharge data is based primarily on measurement and is publically available over the period considered in this review, and their power history is also in the public domain.

Limited data is available from the current fleet of ABWRs as they have been subject to plant shutdowns by the restrictions due to the fukushima daiichi accident in Japan, as such it is not possible to use recent performance data from this generation of reactors. In addition, there are also limitations in the published radiological discharge data for ABWRs. Therefore ABWRs have not been included in this review.

As well as the BWRs the performance data for two UK reactors, Sizewell B and Heysham 2 has also been included. It is considered that the inclusion of these two UK reactors, although not BWRs, will enable a comparison of the UK ABWR performance with latest nuclear power station types operating within the UK.

A summary of the comparative discharges of the UK ABWR and the other stations is presented in Tables 9.5-2 and 9.5-3. The tables report the average discharge in Bq per GWeh over the period 2005 to 2014 inclusive, except for Leibstadt and the US BWRs where performance data is available for 2010 to 2013 inclusive. For comparison, the expected discharges of UK ABWR is based on not the proposed limit but annual discharges and the maximum power of the UK ABWR can be determined by multiplying 1.35 GW by 8760 hours, giving 11826 GWh.

9.5.1. Comparison with Gaseous Discharges

Data has been extracted from the RADD for gaseous discharges for the selected nuclear power stations. The discharge data has been normalised to Bq per GWeh based on the power generation data available on the PRIS database. The resulting performance data is summarised in Table 9.5-2. Generally “total beta/gamma” refers to particulate material in the case of gaseous discharges. Typically these comprise of activated corrosion products such as Co-60. The exact nature of “total beta/gamma” will be defined in the measurement methodologies defined in the site specific permit granted under the Environmental Permitting Regulations. The total beta/gamma of the UK ABWR is equal to the particulates in Table 9.2-1.

Table 9.5-2: Performance Data for BWRs and Expected Annual Discharges for the UK ABWR (Gaseous Discharge)

NPP	Total Noble Gases (Bq per GWeh)	Total Iodine (Bq per GWeh)	Total Beta/Gamma (Bq per GWeh)	H-3 (Bq per GWeh)	C-14 (Bq per GWeh)
UK ABWR	1.5E+08	2.7E+04	2.1E+01	2.3E+08	7.7E+07
Olkiltuoto	4.0E+08	3.4E+03	1.5E+03	2.7E+07	5.7E+07
Gundrennigen B+C	1.9E+08	2.7E+03	1.9E+01	3.6E+07	4.1E+07
Isar 1*	3.3E+08	4.5E+03	Not reported	1.7E+07	5.0E+07
Philippsburg 1*	2.7E+08	7.8E+03	3.8E+03	7.1E+06	6.1E+07
Cofrentes	2.3E+09	1.1E+06	6.5E+06	1.1E+08	5.2E+07
Santa Maria de Garona	3.2E+09	4.6E+05	6.7E+04	3.1E+08	6.0E+07
Forsmark	3.9E+08	1.2E+04	9.2E+03	3.8E+07	9.1E+07
Oskershamn	8.1E+08	3.2E+04	2.0E+05	5.1E+07	4.8E+07
Ringhals 1	7.3E+08	2.8E+04	8.1E+06	3.0E+07	7.8E+07
Leibstadt	5.9E+07	1.0E+04	1.1E+03	2.0E+08	7.1E+07
Clinton-1	1.8E+07	7.3E+01	4.5E+02	8.8E+07	6.5E+07
Grand Gulf-1	3.1E+09	4.5E+03	5.9E+02	1.1E+08	4.0E+07
LaSalle county 1&2	4.6E+09	8.8E+04	2.7E+04	5.0E+07	5.8E+07
Limeric-1&2	2.1E+08	1.7E+02	2.0E+03	9.7E+07	6.1E+07
Nine Mile Point-2	9.1E+08	1.1E+04	1.6E+04	2.9E+08	6.7E+07
Perry-1	1.8E+08	6.6E+02	2.6E+01	2.2E+07	6.3E+07
Susquehanna-1&2	1.6E+07	< MDC**	4.0E+02	4.5E+07	6.6E+07
Sizewell B PWR	4.0E+08	1.0E+04	1.4E+03	1.0E+08	2.9E+07
Heysham 2 AGR	1.6E+09	8.6E+03	1.2E+03	1.2E+08	1.7E+08

* power generation terminated in 2011

** MDC = minimum detectable concentration (used in Susquehanna-1 & -2 data reporting).
If a radionuclide was not detected, zero activity was used for that isotope in dose calculations and the activity is listed as "<MDC" (less than the minimum detectable concentration). <MDC indicates that no activity was positively detected in any sample when samples were analysed with techniques which achieved the required Lower Limits of Detection (LLD).

The cells shaded green in Table 9.5-2 indicate where the UK ABWR discharge value is below the average value and orange cells indicate where the UK ABWR value is below the maximum value in the range shown in the associated figures. The grey cells indicate data for Sizewell B and Heysham 2.

The data in Table 9.5-2 is represented graphically in Figures 9.5-1 to 9.5-5. These figures show the average normalised discharges reported in Table 9.5-2 as solid bars and the minimum and maximum values for the particular site over the period 2005 to 2014 (with the exception of ISar-1 and Phillippsburg-1 which terminated power generation in 2011) as the error bars. Furthermore, available data from 2010 to 2013 for Leibstadt and the US BWRs is used. The expected annual discharge from the UK ABWR, expressed as Bq per GWeh, is represented by the horizontal orange line.

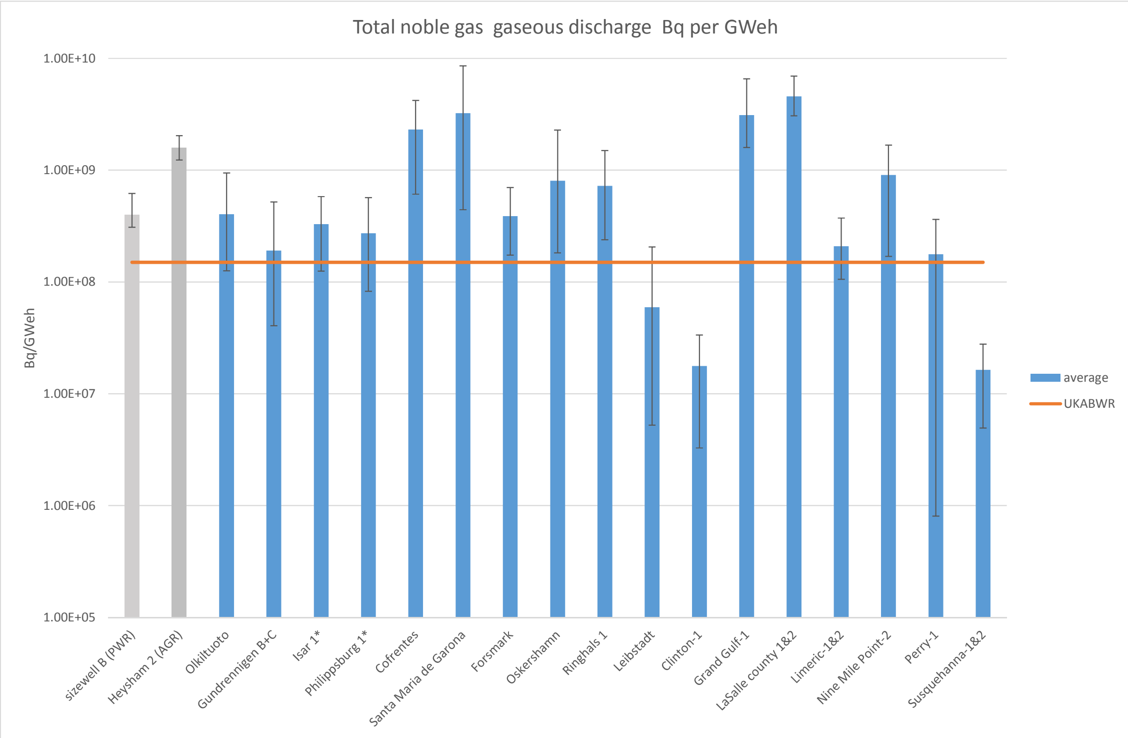

Figure 9.5-1: Total Noble Gas Normalised Annual Gaseous Discharge, Bq per GWeh

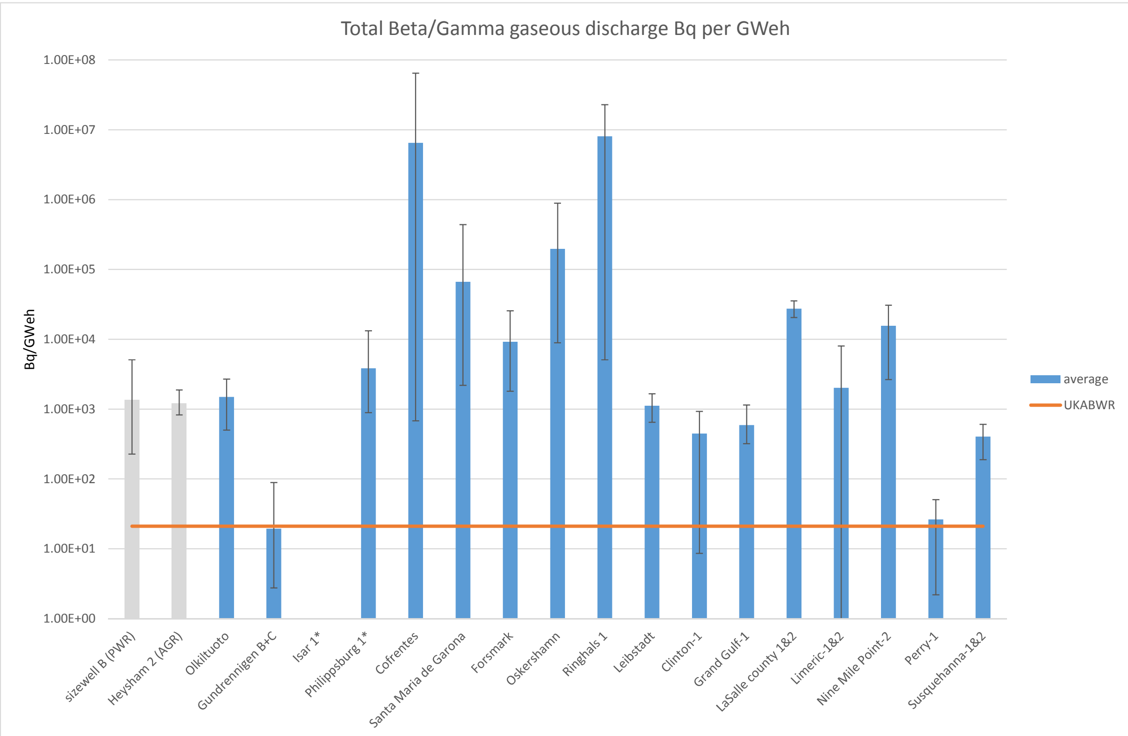

Figure 9.5-2: Total Beta/Gamma Normalised Annual Gaseous Discharge, Bq per GWeh

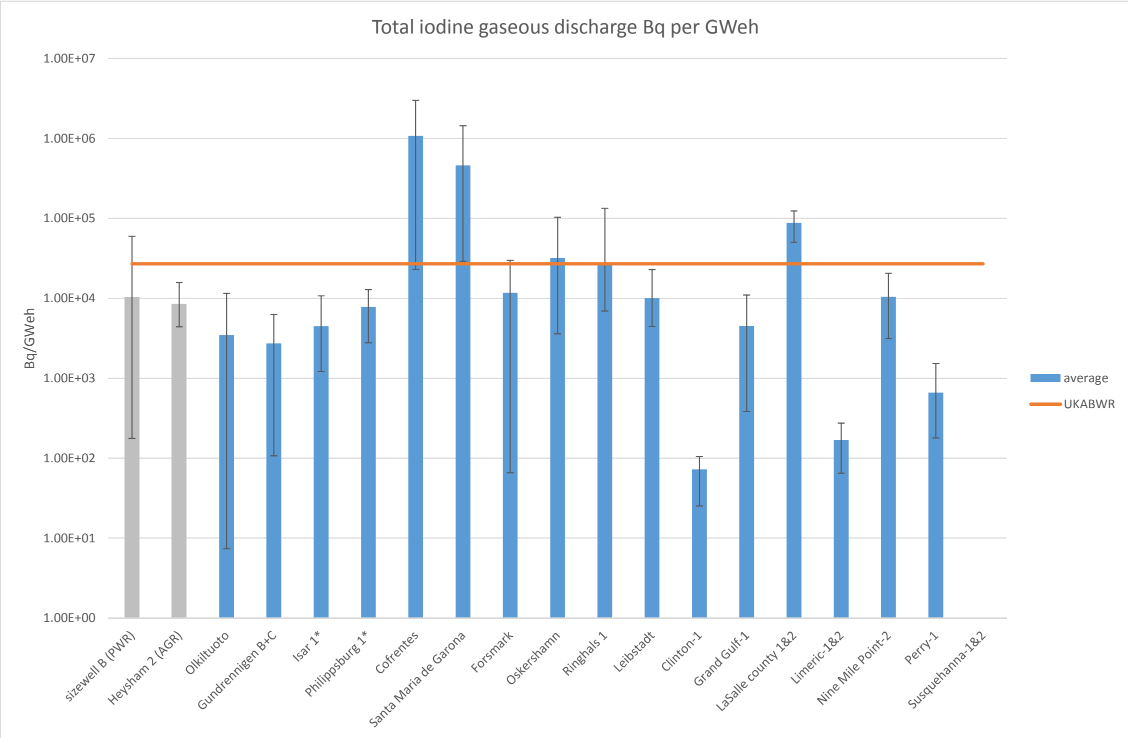

Figure 9.5-3: Total Iodine Normalised Annual Gaseous Discharge, Bq per GWeh

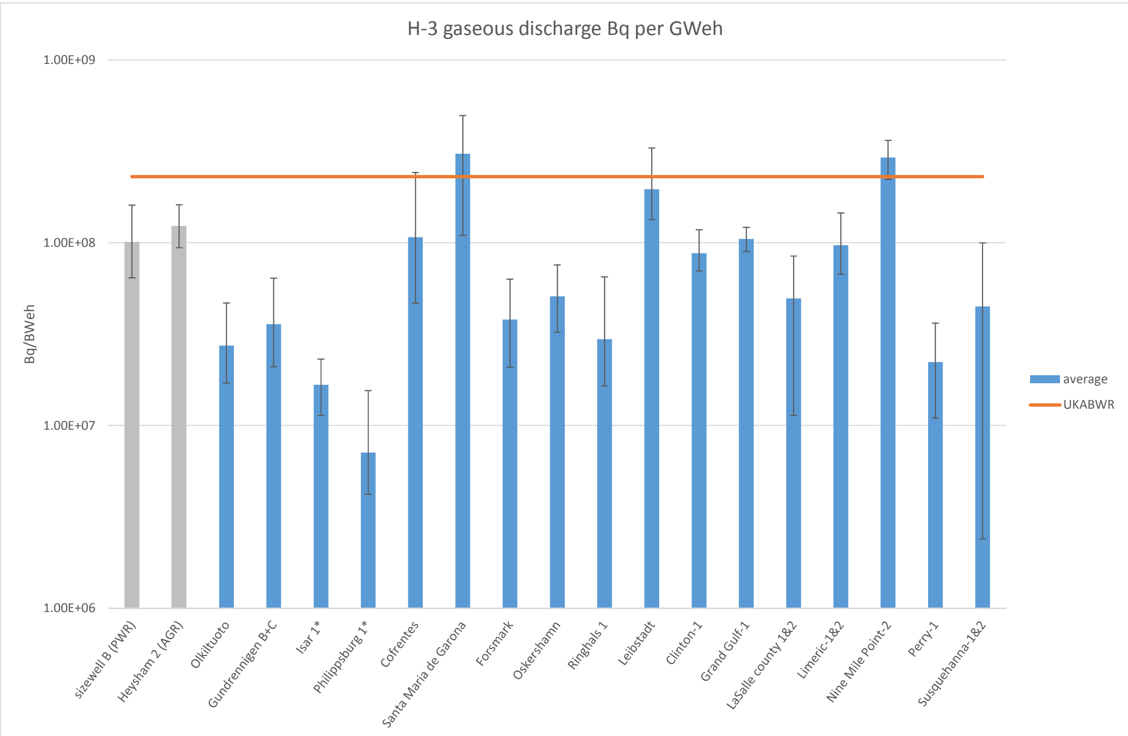

Figure 9.5-4: H-3 Normalised Annual Gaseous Discharge, Bq per GWeh

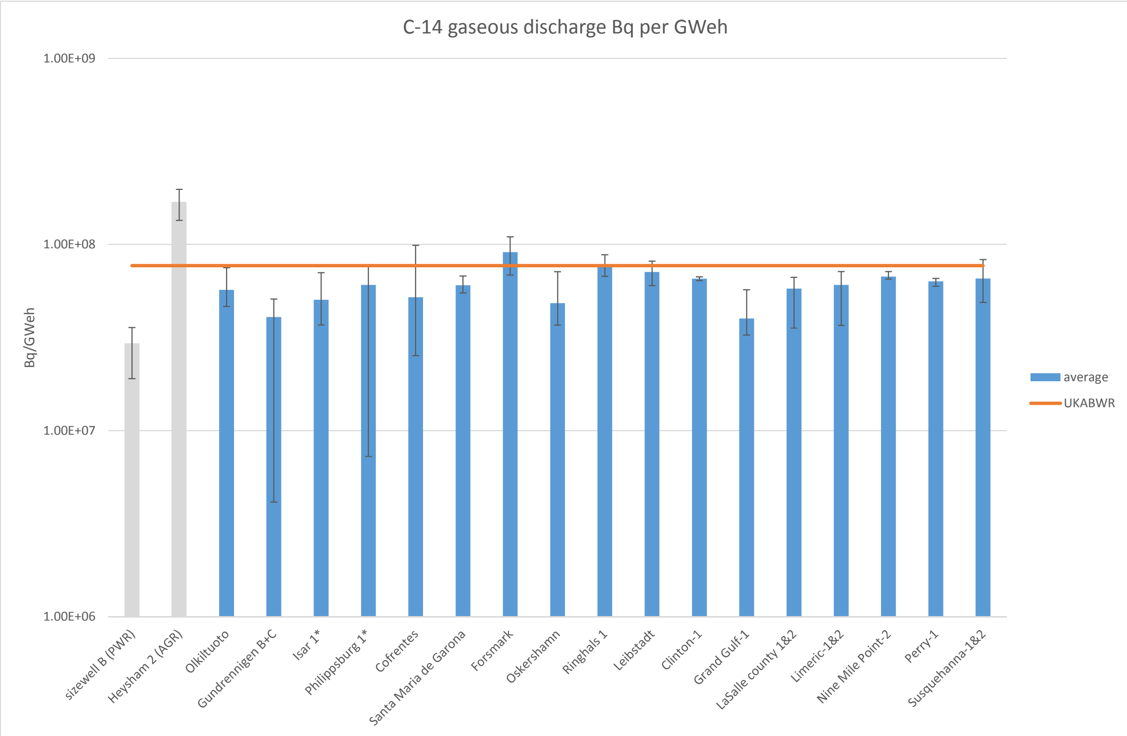

Figure 9.5-5: C-14 Normalised Annual Gaseous Discharge, Bq per GWeh

For the gaseous discharges it can be seen in Figures 9.5-1 and 9.5-2 that the discharges of noble gases and total beta/gamma from the UK ABWR are expected to be lower per GWeh than the majority of the comparison plants. This is considered to be due to the application of BAT throughout the design of the UK ABWR. For example modern fuel is designed to be more robust and less prone to minor cladding defects that may lead to the release of fission products in to the coolant. In addition the OG system proposed for the UK ABWR consists of four all welded activated charcoal beds in series resulting in a marked reduction in the potential for bypassing of the bed due to leakage. The OG system is also cooled which improves the retention of noble gases, hence further reducing the discharge. The gaseous discharges pass though filtration systems that remove particulate material.

Figure 9.5-3 shows that the predicted annual discharge of iodine to be below the average annual normalised discharges for 5 of the comparison plants and below the maximum of the range for another 1 comparison plant. This gives a total of 6 out of the 17 plants included in the comparison where the UK ABWR normalised discharge is below the mean or maximum normalised annual discharge. The seemingly higher annual discharges of iodine predicted to arise from the UK ABWR can be attributed to conservative assumptions made about the decontamination factor of HEPA filter (DF=1). Some of the iodine species are expected to occur in the particulate form, the majority of which will be filtered prior to discharge.

The predicted annual gaseous discharge of H-3 is predicted to be below the average annual normalised discharge presented in Figure 9.5-4 for 2 of the comparison plant and below the maximum of the range for a further 2 of the comparison plant. This gives a total of 4 out of the 17 plants included in the comparison where the UK ABWR normalised discharge is below the mean or maximum normalised annual discharge. The UK ABWR normalised H-3 discharge may exceed the other 13 BWRs normalised discharges due to pessimisms in the calculation of H-3 discharges via the TGS system and the HVAC system. The majority of the H-3 discharge comes from the steam flow of the TGS system and evaporation at the SFP via the HVAC system. The steam flow rate of the TGS system is set by including some conceivable maximum margin compared to actual flow rate in plant operation. The evaporation rate of SFP is also sensitive to the SFP water temperature. A reduction in the assumed water temperature by 2°C will result in an approximately 15% reduction in the discharge of H-3 from the SFP.

The predicted annual gaseous discharge of C-14 are below the average annual normalised discharge presented in Figure 9.5-5 for 2 of the comparison plant and below the maximum of the range for a further 4 of the comparison plant. This gives a total of 6 out of the 17 plants included in the comparison where the UK ABWR normalised discharge is below the mean or maximum normalised annual discharge. Unlike the annual discharges from the European nuclear power plant which are based on measurement, the majority of the annual discharges of C-14 from the US nuclear power plant are based on the EPRI methodology which is a calculated value derived from the reactor power.

It is considered that the performance of the UK ABWR is not significantly better than currently operating BWRs because no viable abatement technique is currently available to abate H-3 and C-14; the assessment of techniques for their removal from gaseous wastes presented in [Ref-15] indicates that a number of techniques exist but none has thus far been on operational reactors. In addition to this, IAEA Technical report No.421 concluded that methods for the separation of C-14 and H-3 from gaseous wastes are costly and require high energy consumption and that application of these separation technologies may therefore be limited by their high cost. H-3 that may be present in the OG system is removed from the gas stream after passing through the recombiner and OG system condenser. This is because the recombiner causes free hydrogen in the gas stream to react with oxygen to form water which is then condensed into the liquid phase. Since H-3 displays the same chemical properties as standard hydrogen the H-3 also follows this route thus removing it from the gas stream. As a result of the engineered controls provided on the OG system the primary source of H-3 discharge to air is via the TGS system and the HVAC system. It has been demonstrated in the Demonstration of BAT report that the cost (in terms of time, trouble and effort) of further optimising the UK ABWR in order to minimise discharges of H-3 (including the provision of H-3 abatement) is grossly disproportionate compared to the potential benefits in terms of dose reduction. C-14 will be discharged via the OG system and it has been demonstrated to be grossly disproportionate to provide abatement [Ref-15].

It can be seen from the above figures that with the exception of Iodine and H-3 from Heysham 2, and H-3 and C-14 from Sizewell B, the expected discharges from the UK ABWR are lower than the performance data for the two reactor types currently operating in the UK. These reactor types are the PWR operating at Sizewell B and the latest version of the AGR operating at Heysham 2.

9.5.2. Comparison with Liquid Discharges

Data has been extracted from the RADD for liquid discharges for the nuclear power stations considered in this comparison. The discharge data has been normalised to Bq per GWeh based on the power generation data available on the PRIS database. The resulting performance data is summarised in Table 9.5-3. For liquid discharges total beta/gamma refers to all detectable radionuclides excluding H-3 and any other radionuclide that may be subject to a separate limit. The exact nature of “total beta/gamma” will be defined in the measurement methodologies defined in the site specific permit granted under the Environmental Permitting Regulations. The total beta/gamma of the UK ABWR is equal to other radionuclides excluding H-3 in Table 9.2-2.

Table 9.5-3: Performance Data for BWRs and Expected Annual Discharges for the UK ABWR (Liquid Discharge)

NPP	H-3 (Bq per GWeh)	Total Beta/Gamma (Bq per GWeh)
UKABWR	1.7E+07	2.6E+02
Olkiltuoto	1.3E+08	2.2E+04
Gundrennigen B+C	1.8E+08	4.5E+04
Isar 1*	8.2E+07	9.8E+03
Philippsburg 1*	9.7E+07	2.8E+04
Cofrentes	6.5E+07	1.6E+04
Santa Maria de Garona	1.4E+08	9.1E+04
Forsmark	8.1E+07	7.8E+03
Oskershamn	8.4E+07	1.4E+05
Ringhals 1	1.3E+08	5.3E+05
Leibstadt	2.5E+08	1.4+E04
Clinton-1	NR**	NR
Grand Gulf-1	3.1E+08	1.9E+05
LaSalle county 1&2	<LLD***	<LLD
Limeric-1&2	2.3E+07	2.8E+03
Nine Mile Point-2 (2010 data only)	2.0E+07	1.1E+03
Perry-1	1.2E+08	1.8E+05
Susquehanna-1&2	1.4E+08	1.1E+05
Sizewell B	5.0E+09	1.2E+06
Heysham 2	3.7E+10	7.4E+06

* power generation terminated in 2011
** NR = No Release i.e. no releases occurred during this period.
***LLD = Lower Limit of Detection. No radioactivity was detected and represents the lower limit of detection value for samples within a data set.

The green cells in the table above indicate where the UK ABWR value is below the average value and orange cells indicate where the UK ABWR value is below the maximum value in the range shown in the associated figures.

The above data are represented graphically in Figures 9.5-6 and 9.5-7 below for the normalised H-3 and total beta/gamma discharges, respectively. These figures show the average normalised discharge reported in Table 9.5-3 as solid bars and the minimum and maximum values for the particular site over the period 2005 to 2014 (with the exception of ISar-1 and Phillippsburg-1 which terminated power generation in 2011) as the error bars. Furthermore, available data from 2010 to 2013 for Leibstadt and the US BWRs is used. The expected annual discharge from the UK ABWR, expressed as Bq per GWeh, is represented by the horizontal orange line.

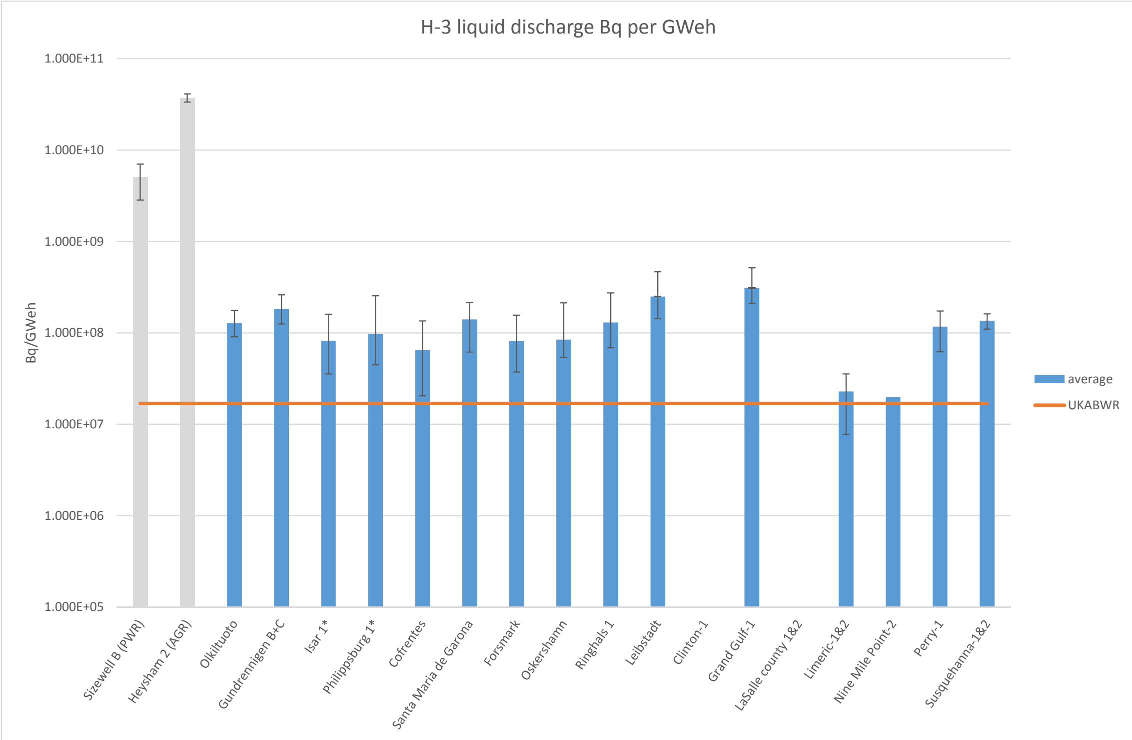

Figure 9.5-6: H-3 Normalised Annual Liquid Discharge, Bq per GWeh

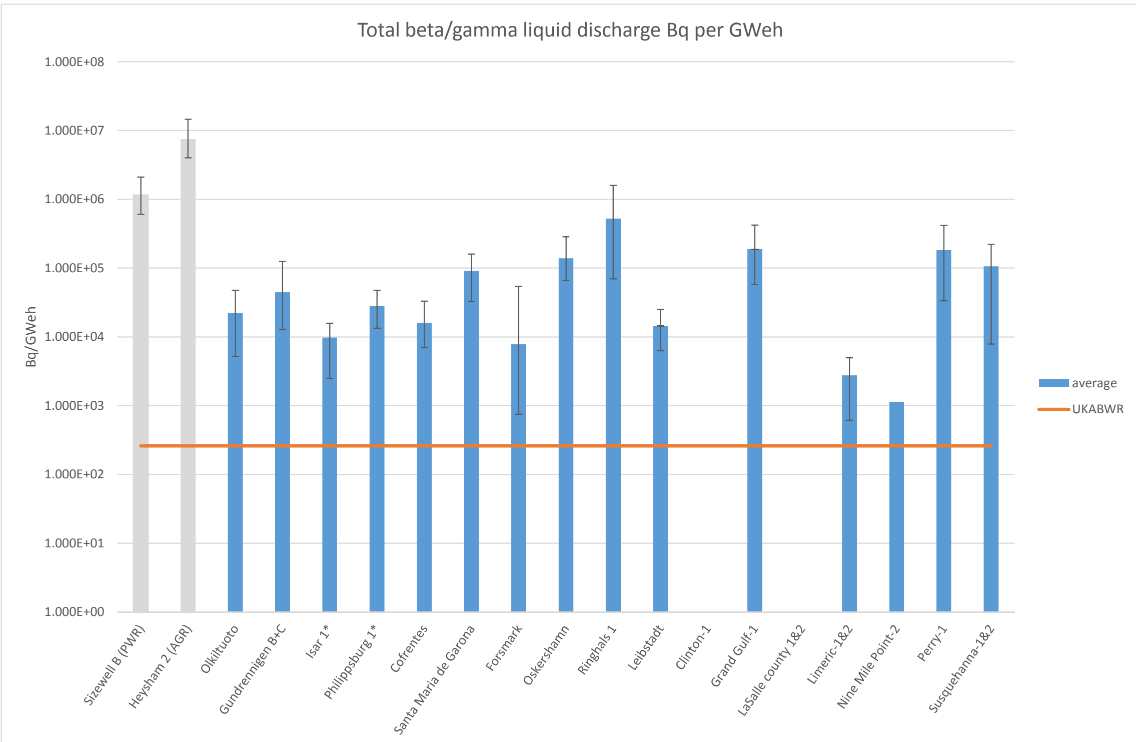

Figure 9.5-7: Total Beta/Gamma Normalised Annual Liquid Discharge, Bq per GWeh

For the liquid discharges the predicted normalised annual discharges from the UK ABWR are lower than the average normalised discharges from the comparison plant for both H-3 and total beta/gamma. This is considered to be due to the application of BAT throughout the design process. BAT has been employed to minimise the amount of activated corrosion products that may be generated. The design of the liquid treatment system includes filters and demineralisers to remove radioactivity from liquid waste streams. In addition the discharges from the HCW and LD sample tanks are sampled and monitored prior to discharge and in the event that radioactivity is found in excess of permitted values the liquid is sentenced for further treatment in the clean-up system before it is discharged.

10. Conclusion

The proposed annual limits presented in this report are based upon the validated and justified discharges of the UK ABWR under normal operating conditions (accounting for expected events) [Ref-6][Ref-7] [Ref-11][Ref-30] [Ref-31] and the headroom factors presented in this document.

The expected performance of the UK ABWR process and abatement systems have been based on OPEX from similar systems, manufacturers' performance data, and the implementation of validated models that describe decontamination factors [Ref-30][Ref-31] – these have all been subject to detailed review. The implementation of BAT across the UK ABWR design has been demonstrated in [Ref-15]. It is considered that a robust justification has been made for the derivation of activity concentrations in reactor water, reactor steam and the performance of process and abatement plant of the UK ABWR. It is therefore also considered that the annual discharge data, for normal operation, presented in this report have been justified.

The joint regulatory guidance 'Principles for the Assessment of Prospective Public Doses arising from Authorised Discharges of Radioactive Waste to the Environment' [Ref-28] describes the various dose limits and dose constraints that apply to prospective dose assessments for members of the public. This information is reproduced here in Appendix A for comparison against the data presented in Table 9.2-1 and Table 9.2-2, which include an indicative estimate of the annual dose to a member of the public from the discharge to the environment for each radionuclide at the proposed limits.

The overall indicative annual exposures of a member of the public due to gaseous and liquid discharges during normal operation of a single UK ABWR unit are estimated to be approximately 24 µSv/y and 0.001 µSv/y, respectively. These estimates are based on the discharges presented in Table 9.2-1 and Table 9.2-2 of this report. The direct dose contribution is approximately 1 µSv/y, resulting in a summated annual exposure of about 25µSv/y [Ref-12].

These radiological dose estimates are well below the legal limit of 1mSv per year and the source constraint of 0.3mSv per year.

Appendix A: UK Dose Limits & Constraints Taken from [Ref-28]

Effective dose criteria	**Dose quantity**	**Application to prospective dose assessments**	**Purpose of assessment**
Dose limit	1 mSv/y	One or more future discharges are planned and the radioactivity will combine with the residues of past discharges from one or more sources and direct radiation.	To show that total doses from one or more past and present and future sources will not exceed dose limit.
Site constraint	0.5 mSv/y	Future discharges from the planned operation of more than one source where the sources are on sites that are adjacent. Direct radiation is not included.	To assist optimisation of the planned operation of sources where the sources are under separate control but located close together.
Source constraint	0.3 mSv/y	Upper constraint on future discharges and direct radiation from the planned operation of a single source. Dose assessment should be refined until it is considered to be sufficiently realistic or falls below 0.02 mSv/y.	To assist constrained optimisation of the planned operation of a single source. Provide a realistic assessment of doses to act as an input to the optimisation process.
Between dose threshold and source constraint	0.02 to 0.3 mSv/y	Future discharges and direct radiation from the planned operation of a single source. Dose assessment should be refined until it is considered to be sufficiently realistic or the assessed dose falls below 0.02 mSv/y.	To assist constrained optimisation of the planned operation of a single source Provide a realistic assessment of doses to act as an input to the optimisation process.
Level of dose below which the dose assessment requires no further work.	0.02 mSv/y	Future discharges and direct radiation from the planned operation of a single source. If doses are below this threshold, the dose assessment need not be refined further.	To assist constrained optimisation of the planned operation of a single source. Sufficiently low doses to be used as an input to the optimisation process without further refinement.

Appendix B: Schematic Drawings Showing the Process Source Term Calculation Points Used in the Discharges Assessments

The following diagrams are taken from [Ref-31], and are shown here for convenience and easy reference only. The Figures are as follows:

Figure B1: PrST for OG System

Figure B2: PrST for FPC System

Figure B3: PrST for Suppression Pool

Figure B4: PrST for Condensate System

Figure B5: PrST for RD and LCW Systems

Figure B6: PrST for RD, HCW, LD and CONW Systems

Figure B7: PrST for SS System

Figure B8: PrST for TGS System and CST

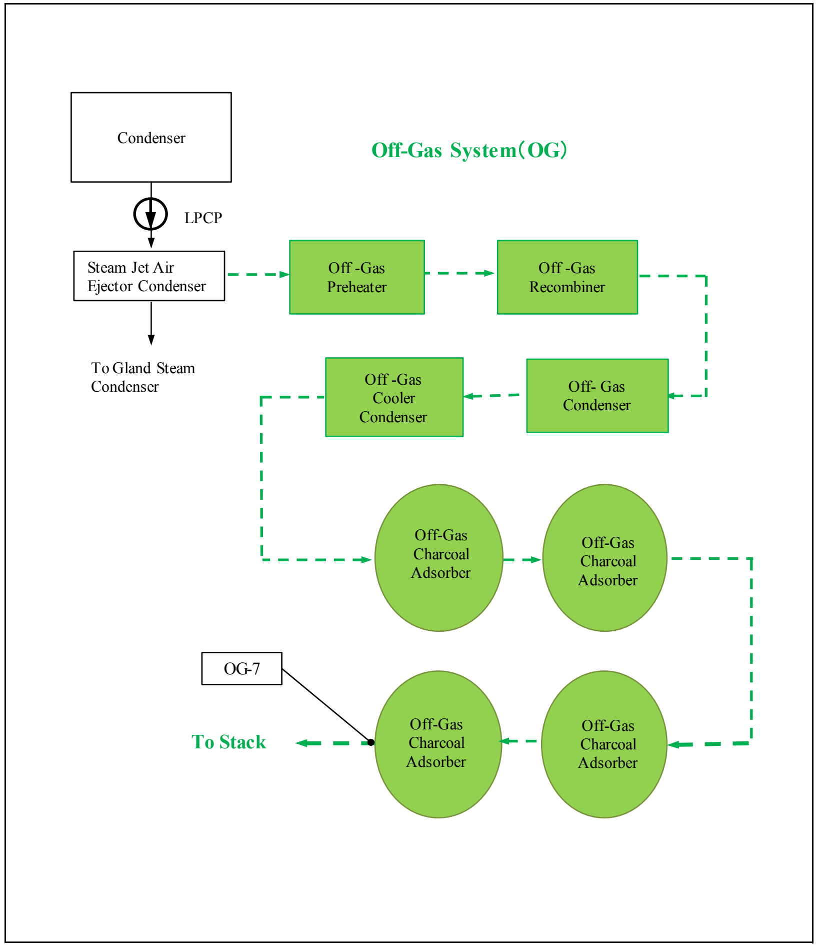

Figure B1: PrST for OG System

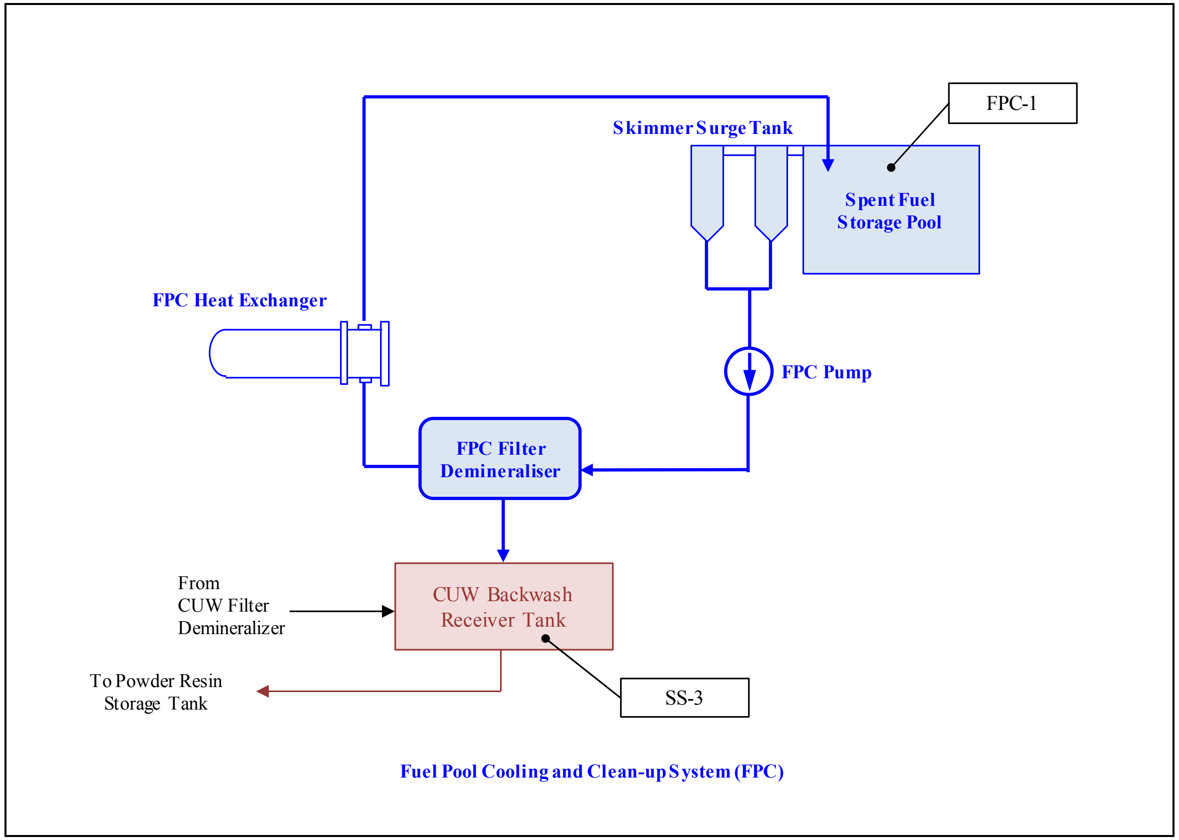

Figure B2: PrST for FPC System

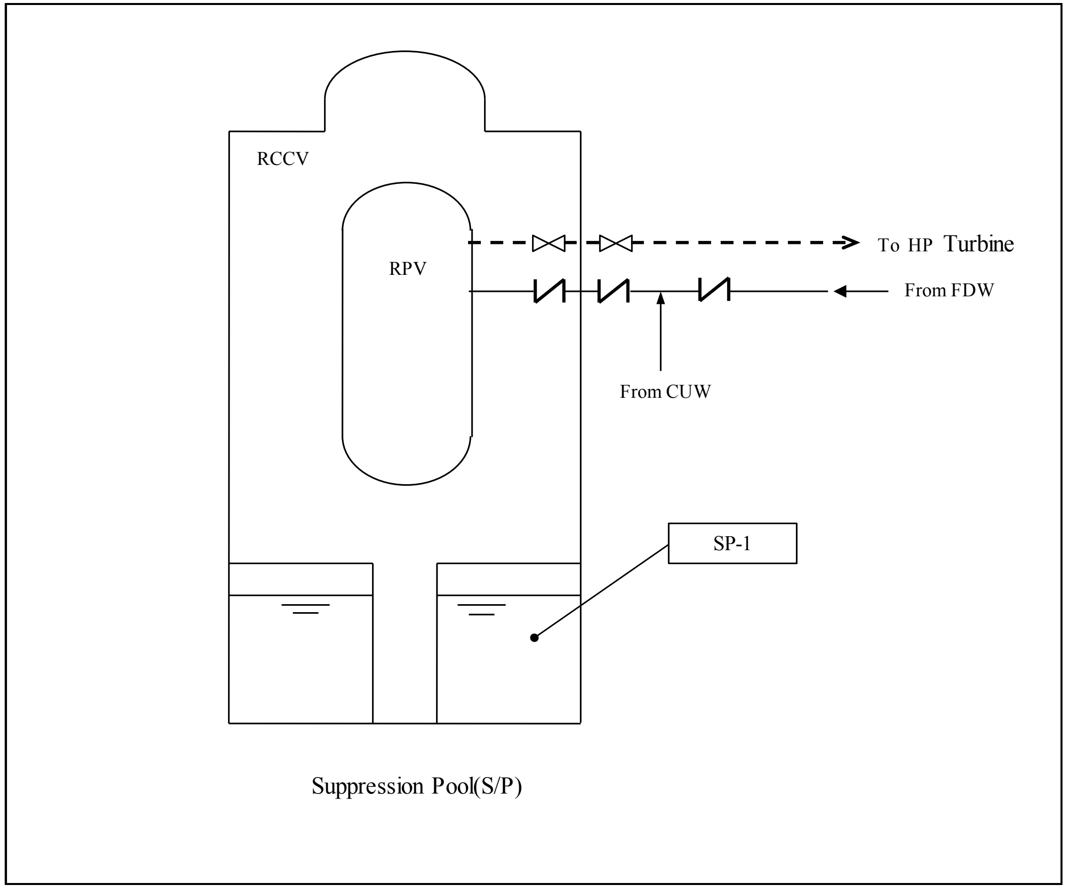

Figure B3: PrST for Suppression Pool

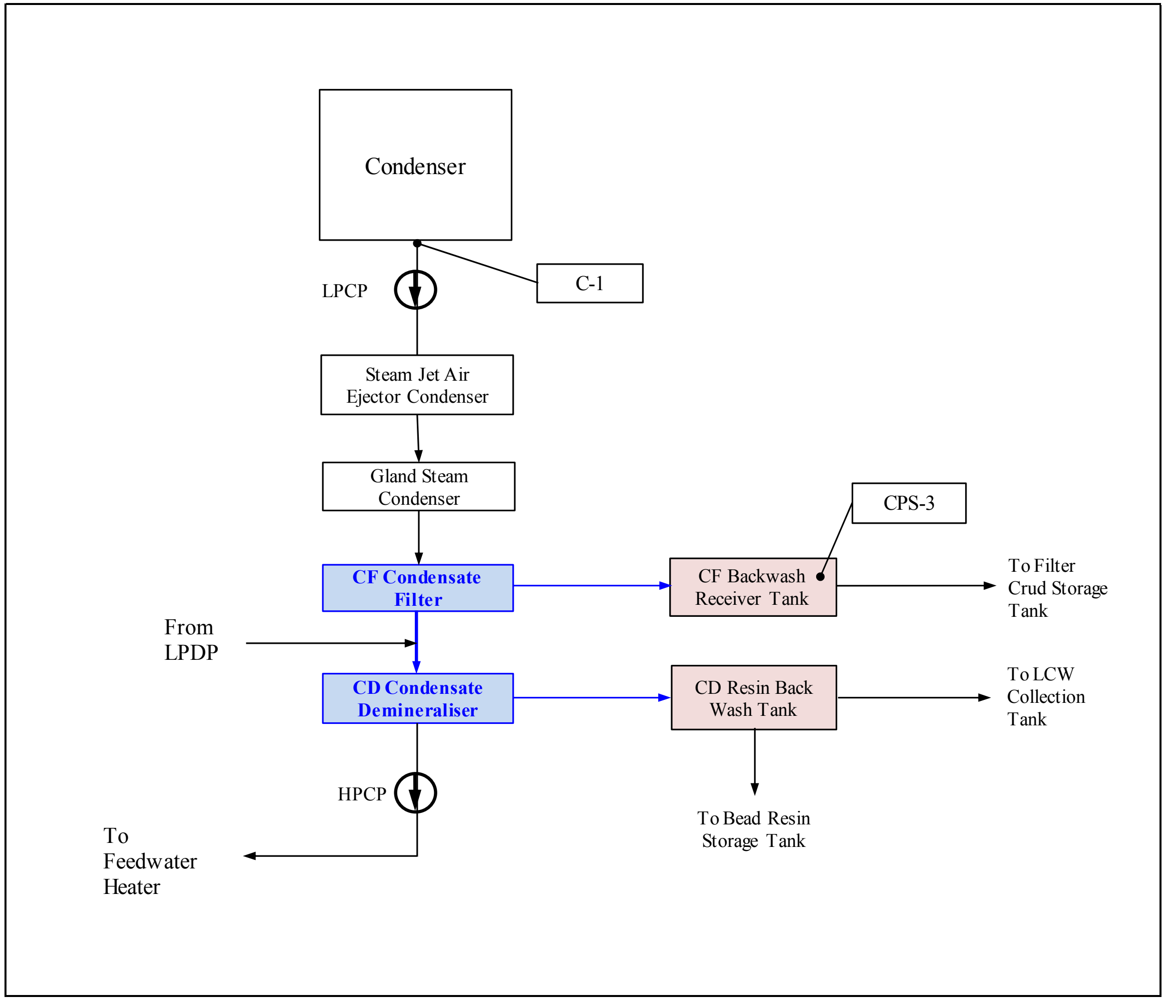

Figure B4: PrST for Condensate System

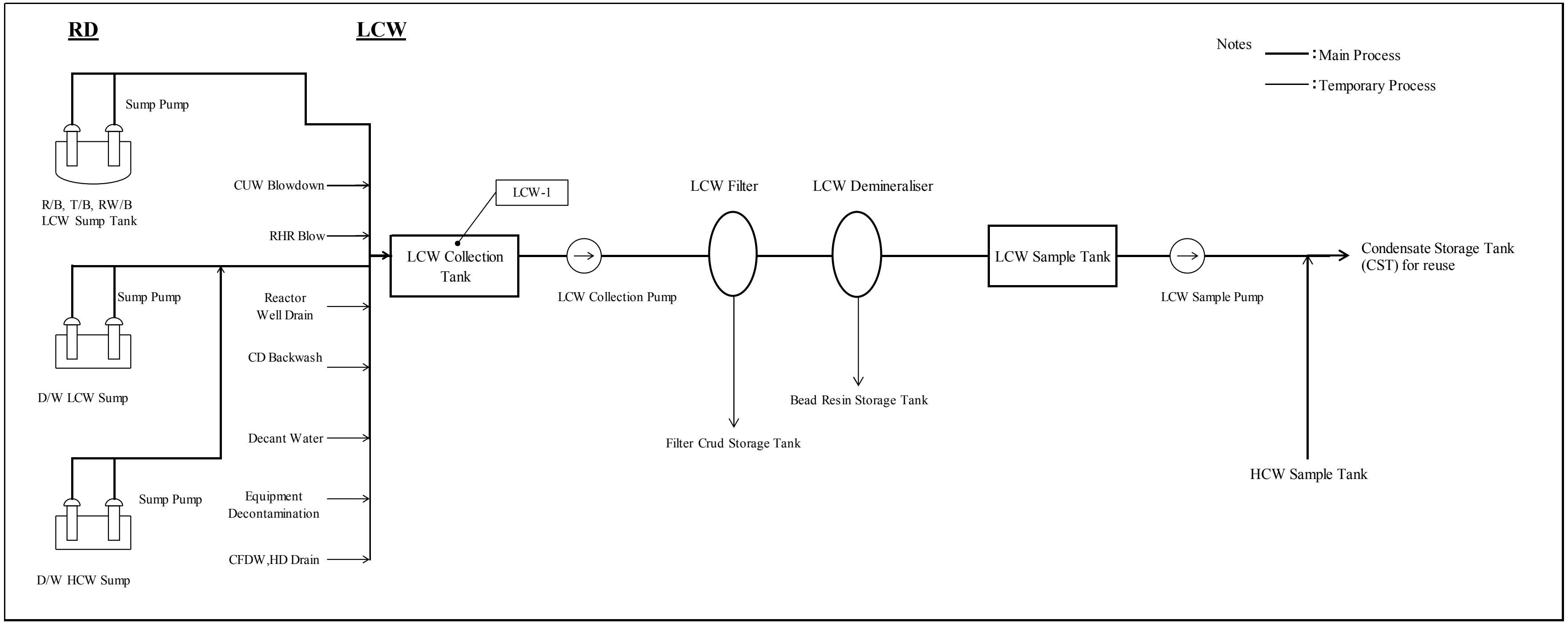

Figure B5: PrST for RD and LCW Systems

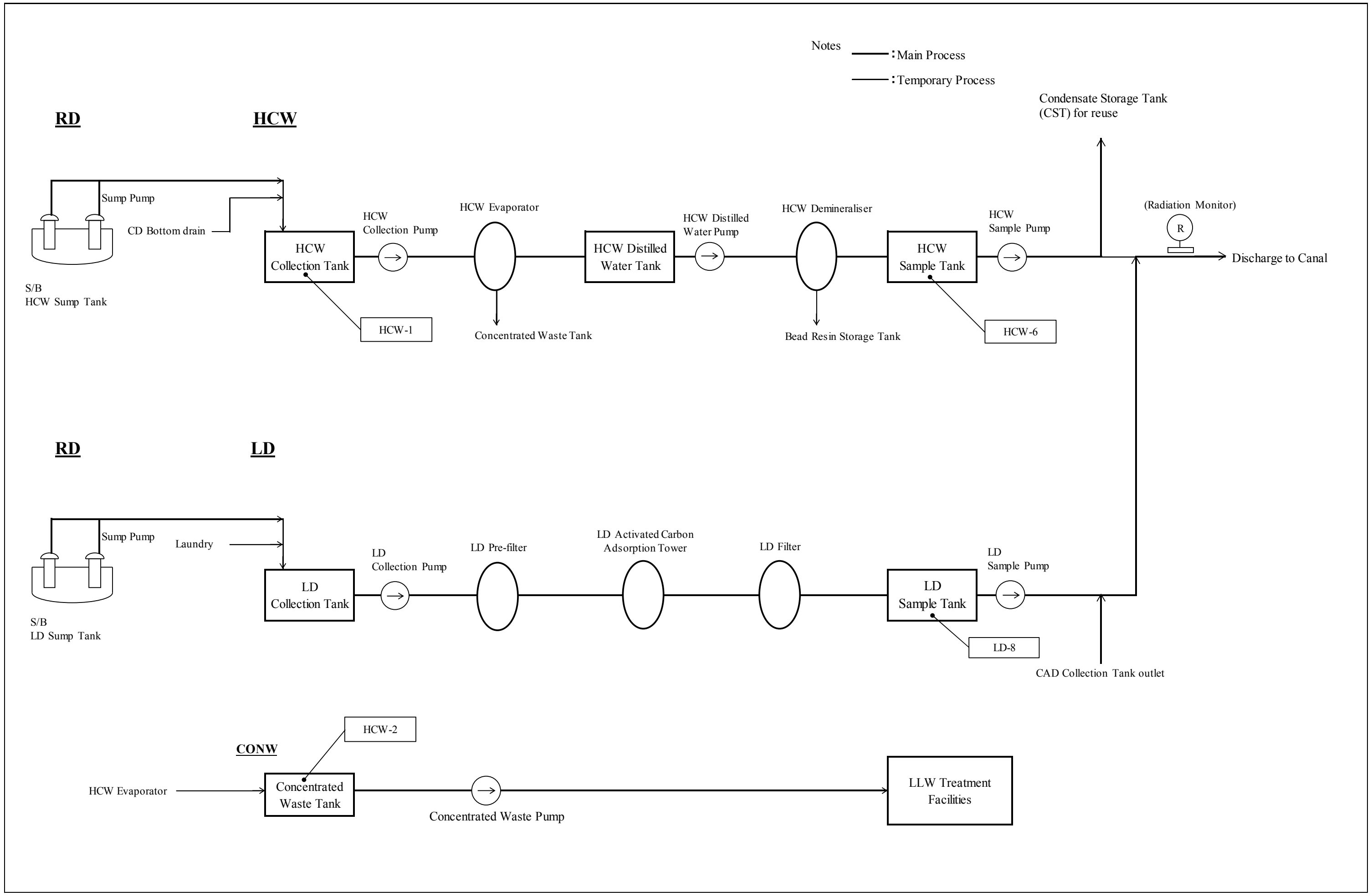

Figure B6: PrST for RD, HCW, LD and CONW Systems

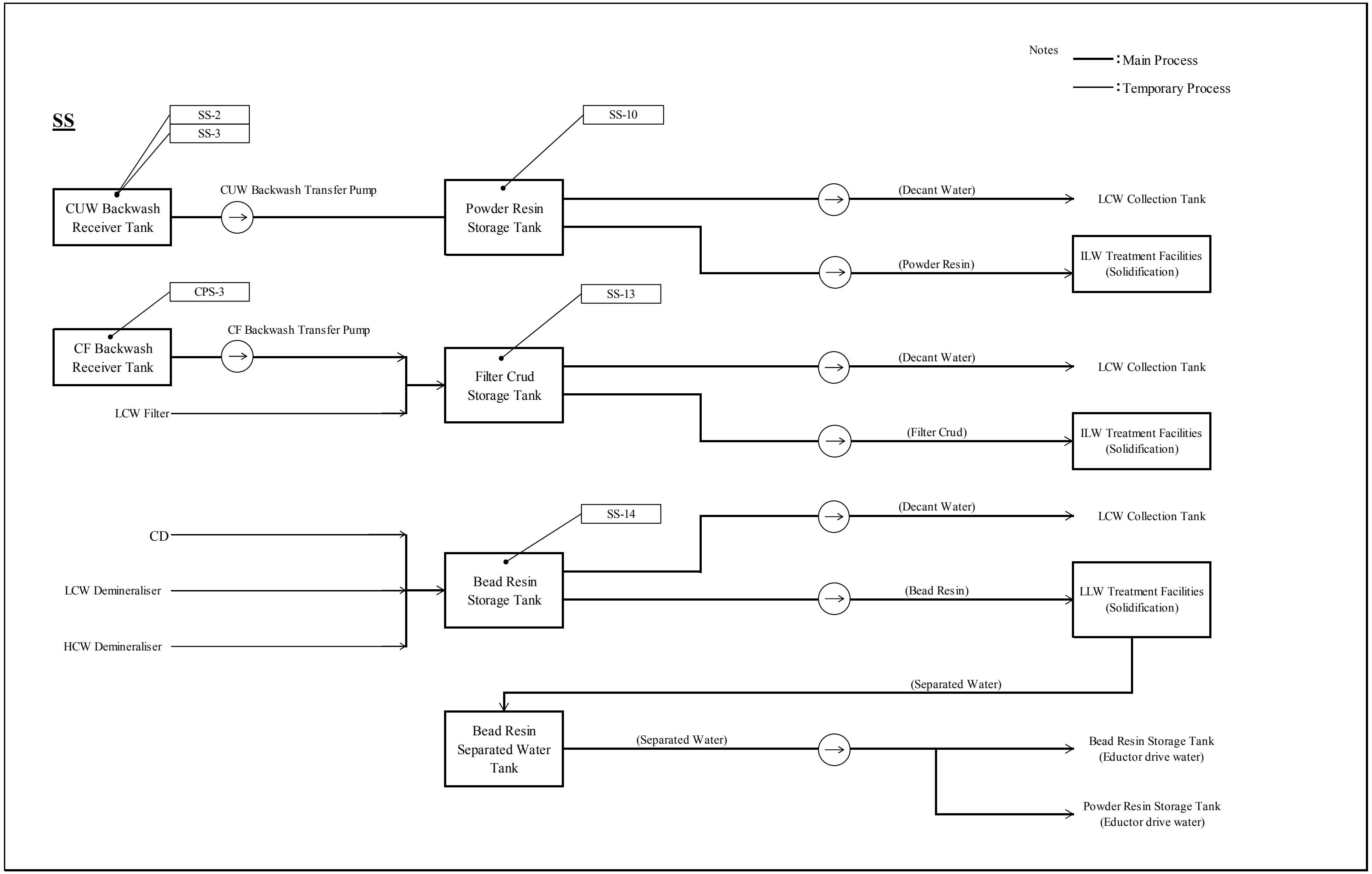

Figure B7: PrST for SS System

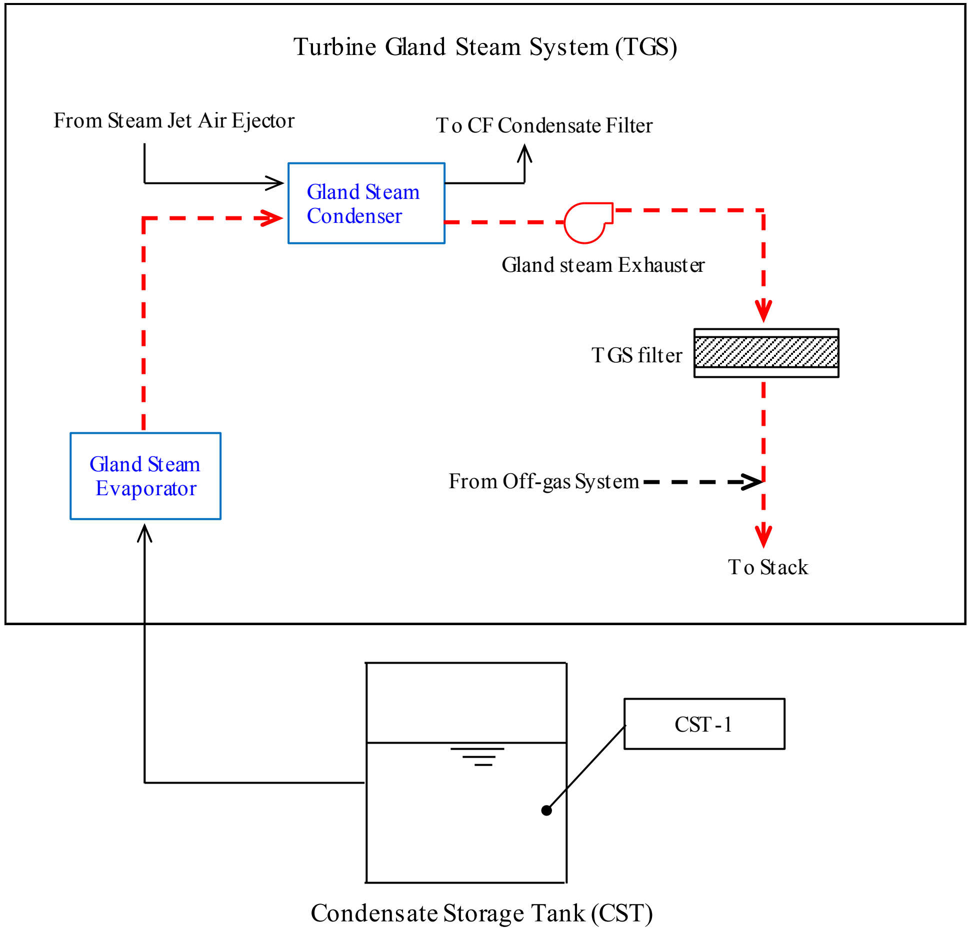

Figure B8: PrST for TGS System and CST