Form10/00

UK ABWR

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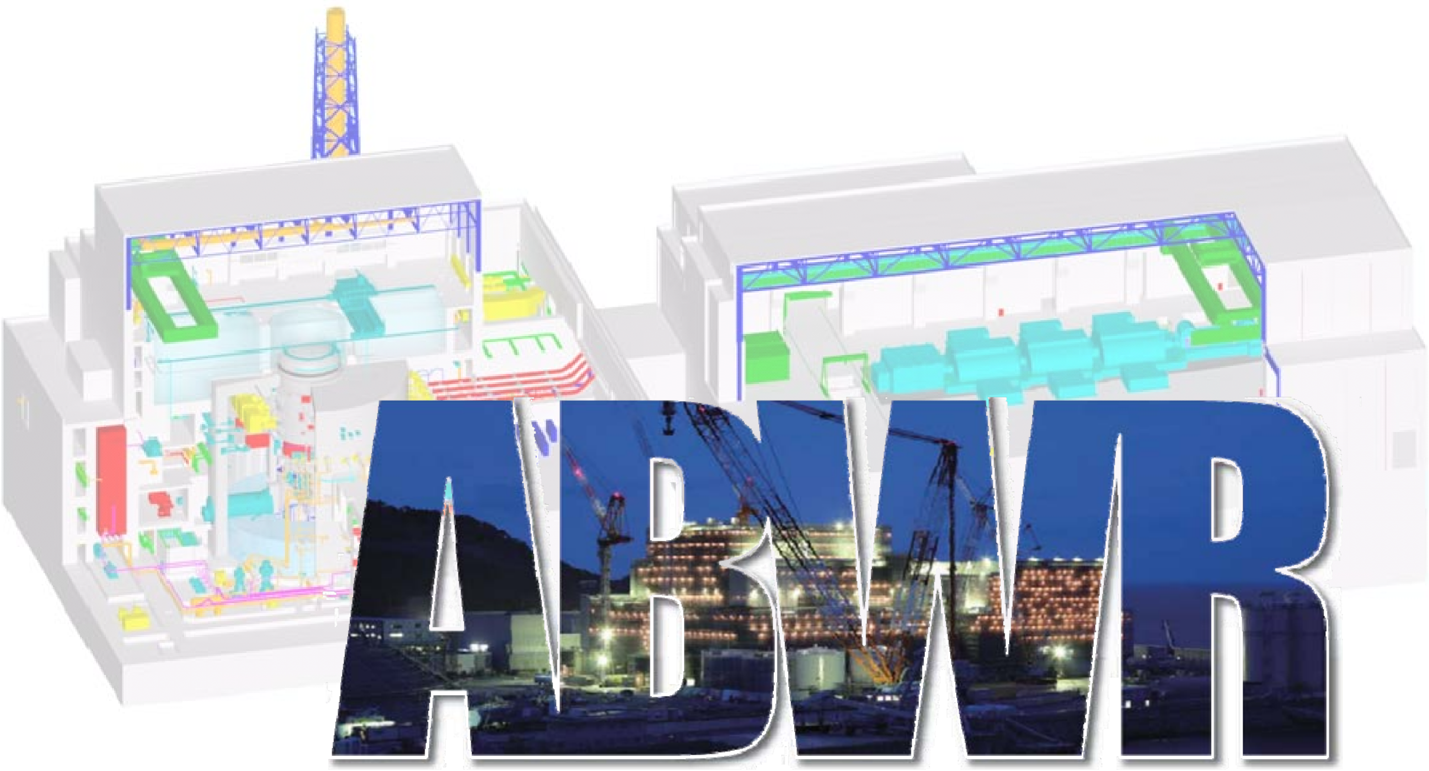
UK ABWR Generic Design Assessment

Quantification of Discharges and Limits



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1. Acronyms

ABWR	Advanced Boiling Water Reactor
AC	Atmospheric Control System
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
BAT	Best Available Technique
BPEO	Best Practicable Environmental Option
BPM	Best Practicable Mean
Bq	Becquerel
BSS	Basis Safety Standards Directive
BWR	Boiling Water Reactor
C&I	Control and Instrumentation
CAD	Controlled Area Drain
CCI	Commercially confidential information
CD	Condensate Demineraliser
CDL	Calculated Detection Limit
CF	Condensate Filter
COMAH	Control of Major Accident Hazards
CONW	Concentrated Waste System
CP	Corrosion Product
CSG	Combustion Sector Guidance Note
CST	Condensate Storage Tank
CUW	Reactor Water Clean-up System
CW	Circulating Water System
CWP	Circulating Water Pump
DORIS	The marine dispersion model used in PC-CREAM 08®
D/W	Dry well
DAW	Dry Active Waste
DCD	Design Control Document
DECC	Department of Energy and Climate Change
DEFRA	Department for Environment, Food and Rural Affairs
DF	Decontamination Factor
DPUR	Dose Per Unit Release

EIA	Environmental Impact Assessment
EMCLs	Environmental Media Concentration Limits
EPR/EPR10	Environmental Permitting (England and Wales) Regulations 2010
EQS	Environment Quality Standards
ERICA	Environmental Risk from Ionising Contaminants: Assessment and Management
ESE	Environmentally Sensitive Equipment
EU	European Union
f-value	Fuel leakage rate
F/D	Filter demineraliser
FAP	Forward Action Plan
FDP	Funded Decommissioning Programme
FDW	Feedwater System
FP	Fission Product
FPC	Fuel Pool Cooling and Clean-up System
GDA	Generic Design Assessment
GDF	Geological Disposal Facility
GEP	Generic Environmental Permit
GNF	Global Nuclear Fuel
GSD	Generic Site Description
HAW	Higher Activity Waste
HCEP	How to comply with your environmental permit
HCW	High Conductivity Waste System
HEPA	High Efficiency Particulate Air Filter
HFE	Human Factors Engineering
HFF	Hollow Fibre Filter
HLW	High Level Waste
HNCW	HVAC Normal Cooling Water System
HOP	Hydrazine, Oxalic acid, Potassium permanganate
HS	Heating Steam System
HSCR	Heating Steam and Condensate Water Return System
HSD	Hot Shower Drain
HSE	Health and Safety Executive (UK)
HVAC	Heating Ventilation and Air Conditioning System
HWC	Hydrogen Water Chemistry
I&C	Instrumentation and Control

IA	Instrument Air System
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEX	Ion-exchange (demineraliser) system
ILW	Intermediate Level Waste
IPPC	Integrated Pollution Prevention and Control
IRA	Initial Radiological Assessment
IWS	Integrated Waste Strategy
KK-6	Kashiwazaki-Kariwa Nuclear Power Station Unit 6
KK-7	Kashiwazaki-Kariwa Nuclear Power Station Unit 7
LCW	Low Conductivity Waste System
LD	Laundry Drain System
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LoC	Letter of Compliance
LOCA	Loss of Coolant Accident
LPRM	Local Power Range Neutron Monitor
LS	Laundry System
LWR	Light Water Reactor
MCERTS	Monitoring Certification Scheme
MS	Main Steam System
NDA	Nuclear Decommissioning Authority
NHS	Non Human Species
NMCA	Noble Metal Chemical Addition
NPP	Nuclear Power Plant
NRW	Natural Resources Wales
NUREG	Nuclear Regulatory Commission Regulation (US)
OG	Off-gas
ONR	Office for Nuclear Regulation
OSPAR	Oslo and Paris Convention on Protection of the Marine Environment of the North East Atlantic
P&D	Plumbing and Drainage system
P&ID	Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs
P/C	Power Centre
PCI	Pellet cladding interaction

PCSR	Pre-Construction Safety Report
PI	Personal information
ppb	Parts per billion
PWR	Pressurised Water Reactor
QA	Quality Assurance
QAP	Quality Assurance Plan
QC	Quality Control
QMP	Quality Management Plan
QMS	Quality Management System
R/B	Reactor Building
RCLEA	Radioactively Contaminated Land Exposure Assessment
RCW	Reactor Building Cooling Water System
REP	Radioactive Substances Regulation – Environmental Principle
RGP	Relevant Good Practice
RP	Requesting Party
RPDP	Radiation Protection Developed Principle
RQ	Risk Quotient
RSA	Radioactive Substances Act
RSR	Radioactive Substances Regulation
RSW	Reactor Building Service Water System
RW/B	Radwaste Building
RWMA	Radioactive Waste Management Arrangement
RWMD	Radioactive Waste Management Directorate
S/B	Service Building
S/P	Suppression Pool
SA	Station Service Air System
SAM	Sampling System
SAP	Safety Assessment Principle
SF	Spent Fuel
SFAIRP	So Far As Is Reasonably Practicable
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SLC	Stand by Liquid Control System
SoDA	Statement of Design Acceptability

SPCU	Suppression Pool Clean-up System
SQEP	Suitably Qualified and Experienced Person (UK)
SRNM	Start-up Range Neutron Monitor
SS	Spent Sludge System
Sv	Sievert
T/B	Turbine Building
TCW	Turbine Building Cooling Water System
TIP	Traversing in-core probe
TSW	Turbine Building Service Water System
TV	Tank Vent Treatment System
UF	Uncertainty Factor
UK	United Kingdom
US	United States
VLLW	Very Low Level Waste
WENRA	Western European Nuclear Regulators' Association

2. References

1 Process and Information Document for the Generic Assessment of Candidate Nuclear Power Plant Designs, Environment Agency, version 2, March 2013.

2 Prospective Dose Modelling, GA91-9901-0026-00001, HE-GD-0005, Rev C, Hitachi-GE, March, 2014.

3 Environmental Permitting (England and Wales) Regulations, SI 2010, No 675, 2010.

4 Statutory Guidance to the Environment Agency concerning the regulation of radioactive discharges to the environment, Department of Energy and Climate Change, 2009.

5 Demonstration of BAT, GA91-9901-0023-00001, XE-GD-0097, Rev C, Hitachi-GE, March, 2014.

6 Regulatory Guidance Series, No RSR 1; Radioactive Substances Regulation – Environmental Principles; Version 2; April 2010.

7 Consideration of and Compliance with the REPs, GA91-9901-0028-00001, XE-GD-0099, Rev C, Hitachi-GE, March, 2014.

8 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.

9 Criteria for setting limits on the discharge of radioactive waste from nuclear sites, (no document number), Environment Agency guidance, 2012.

10 Radioactive Waste Management Arrangements, GA91-9901-0022-00001, WE-GD-0001, Rev C, Hitachi-GE, March, 2014.

11 Regulatory Guide for Evaluating the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities, NSCRG:L- RE-I.01, The Nuclear Safety Commission of Japan, (As Revised), 2001.

12 http://www.tepco.co.jp/nu/kk-np/data_lib/index13-j.html.

13 Sources and effects of ionising radiation, Volume I, Annex B ‘Exposures of the public and workers from various sources of radiation’, UNSCEAR, 2008.

14 Sources and effects of ionising radiation, Annex B ‘Exposures from Man-made Source of Irradiation’, UNSCEAR, 1993.

15 Guide for Radiation Monitoring of Effluent Released from Light Water Nuclear Power Reactor Facilities, NSCRG: L-RE-I.02, Nuclear Safety Commission of Japan, 2001.

16 Internal Hitachi-GE UK ABWR technical document, GE-GD-0001, Hitachi-GE, 2013.

17 Short term releases to the atmosphere, NDAWG-2-2011, National Dose Assessment Working Group Guide, 2011.

18 Developing guidance for setting limits on radioactive discharges to the environment from nuclear licensed sites, SC010034/SR Science Report, Environment Agency, 2005.

19 Principles for the Assessment of Prospective Public Doses arising from Authorised Discharges of Radioactive Waste to the Environment (Joint Regulatory Guidance), Environment Agency, August 2012

3. Introduction

The Environment Agency's requirements for setting proposed discharges and limits as part of GDA are defined within their Process and Information Document (P&ID) (1). This report contains a summary of the initial information and data on the gaseous and liquid discharges anticipated from the normal operation of the UK ABWR, and is line with what has been requested by the Environment Agency in the P&ID.

The initial information within this document has been provided at Step 1b of the GDA process to enable meaningful discussions to take place between Hitachi-GE and the Environment Agency during Step 2. It is expected that the information and data supporting the discharges to the environment will be developed further during GDA Step 2. This in turn will be used to refine the source term and the proposed radioactive discharge limits if appropriate to do so. The results presented in this report establish that, using cautiously derived data and based on initial assessments, the impacts of discharges from the UK ABWR will be within required dose constraints and dose limits. Further information on the associated dose assessment is contained within the Prospective Dose Modelling report (2) which has also been submitted for Environment Agency assessment.

4. Regulatory Context

The Environmental Permitting (England and Wales) Regulations 2010 (EPR10) (3) (as amended) apply limits and constraints on the annual radiation exposure of members of the public. Its principal aims are that the Environment Agency, in exercising its duties and functions under the regulations, ensures that:

- 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 (4) concerning the regulation of radioactive discharges to the environment' 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 reduce further 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 has been assessed in order to demonstrate that BAT has been applied, with the results presented in the Demonstration of BAT report (5). This includes evidence of the following:

- The radioactivity of waste that will be disposed has been minimised;
- The activity of waste discharged has been minimised; and
- Any impacts to members of the public and the environment from such discharges have been minimised.

4.1. REPs Applicable to Discharges and Limit Setting

The methodologies outlined in this report are consistent with industry Relevant Good Practice (RGP) and take into account the relevant Radioactive Substances Regulation – Environmental Principles (REPs) (6). Hitachi-GE's 'Consideration of and Compliance with the Radioactive Substances Regulation Environmental Principles (REPs)' report (7) details the approach undertaken by Hitachi-GE to reviewing and showing compliance with each of the relevant REPs within the GDA submission, highlighting the REPs specifically addressed in each report.

Principally, this report specifically deals with addressing **Principle RSMDP12 '**Limits and levels should be established on the quantities of radioactivity that can be discharged into the environment where these are necessary to secure proper protection of human health and the environment.'

5. Data

5.1. Identification of Radionuclides

The first step in the methodology is the identification of the nuclides to be quantified, to allow definition of the source term.

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 (8), provides the list of radioisotopes to be assessed for liquid or airborne discharges from nuclear power plants. This list differs from the request for the 'identification of the radionuclides to be limited' as expressed in the Environment Agency 'Criteria for setting limits on the discharge of radioactive waste from nuclear sites' June 2012 (9).

The European Commission (2004) document (8) provides a comprehensive list of the radioisotopes to be considered in calculations whilst the Environment Agency guidance (9) is used to determine the radionuclides for which limits will be set. The European Commission document therefore represents a starting point from which the discharge quantification develops into the limit proposal required by the Environment Agency.

The radioisotopes to be considered for releases to atmosphere, as outlined in the EC 2004 document are:

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

The radioisotopes to be considered for liquid releases, as outlined in the EC 2004 document 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	Total-alpha
Co-58	Zr-95	Te-123m	Ba-140	+ Pu-240	H-3
Co-60	Nb-95	Sb-124	La-140	Am-241	

For both gaseous and liquid discharges, Total-alpha should only be reported if nuclide-specific information on alpha-emitters is not available.

It is recognised that the GDA process is a significant undertaking carried out over a number of years, during which time decisions on the design of the UK ABWR will be inevitably made. The associated design details and their effects must be taken into account across the GDA submission. Every opportunity has been taken to ensure the basis of the assessments has been fixed for this early step of GDA, however, if changes occur that do affect the source term they will be taken into account in later submissions. Some potential decisions, such as the introduction of a Hydrogen Water Chemistry (HWC) regime, would not have any material effect on the production of fission or activation products and would not impact upon the assessments carried out, even though their subsequent behaviour (i.e. transportation, deposition) may be altered.

5.2. Mechanisms of formation of the radionuclides

Radioactive species are generated within the UK ABWR via the different mechanisms and processes listed below. These are described in full in the Demonstration of BAT report (5) and other supporting documentation, including the Radioactive Waste Management Arrangements report (10):

- Fission of fuel fissile material;
- Fission of structural Uranium;
- Activation of structural component;
- Activation of coolant substance or impurity;
- Activation of fuel component or impurity;
- Ternary fission in fuel;
- From secondary neutron sources; and
- From Boron in the Control Rods.

Each of these mechanisms will be described in more detail in future submissions of this document and the Prospective Dose Modelling report (2).

5.3. Data sources

There are three types of data that have been used to estimate the source terms for the UK ABWR:

- Calculated values
- Measured actual values
- Assumed values

The derivation of each type of data is outlined in the subsequent three sub-sections of this report.

5.3.1. Calculated values

The gaseous and liquid discharge calculation method is based upon the reference methodology presented in the Japanese Regulatory Guide for Evaluating the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities (11). This calculation is used for estimating the discharges of radioactivity for all light water reactors in Japan and is based on assumed release rates from fuel failures, decontamination factors and activity flow through the off-gas and liquid waste treatment system. The radionuclide concentration is calculated by considering the design basis fuel leakage, flow rate, pressure and decontamination factor at each system. No changes have been made to the established methodology or the values used in the Japanese reference methodology for the UK ABWR Step 1b assessments and they are considered to be appropriate for the UK ABWR at this stage of the GDA process. Further information on the use of this methodology will be provided at Step 2.The gaseous and liquid annual discharge rates are calculated using this methodology and a noble gas release ratio known as the ‘f-value’. The f-value is proportional to the presence of fuel cladding damage that compromises the fuel integrity, allowing fission products to migrate from within the fuel into the cooling water and eventually to the abatement system inlet. Based on operational experience from BWR’s in Japan and expertise in Hitachi-GE, the best performance data (the realistic value) is f=3.7E+7 Bq/s (based on a 30 min decay value over the course of the migration path). This value reflects the continuous improvement seen in the design of both the reactor unit and fuel since the design reference methodology was introduced in 1975. This has significantly reduced the frequency and magnitude of releases resulting from damage to the fuel cladding.

The liquid and gaseous annual discharge rates for each radionuclide calculated using this methodology are shown in Table 5.3-1 and Table 5.3-2.

Table 5.3-1: Annual liquid discharge rate (Calculated)

Radionuclide[2]	Annual discharge (Bq/y)
Cr-51	1.6E+06
Mn-54	3.3E+07
Fe-59	5.7E+06
Co-58	2.5E+06
Co-60	2.5E+07
Sr-89	1.6E+06
Sr-90	8.2E+05
I-131	1.6E+06
Cs-134	4.1E+06
Cs-137	6.5E+06

[2] The radioactive nuclide composition is based on the Japanese Regulatory Guide for Evaluating the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities (11).

Table 5.3-2: Annual gaseous discharge rate (Calculated)

Radionuclide	Annual discharge[1] (Bq/y)
Kr-90	5.3E-05
Xe-139	6.0E-02
Kr-89	2.2E+09
Xe-137	8.1E+09
Xe-135m	1.4E+11
Xe-138	2.6E+11
Kr-87	1.8E+11
Kr-83m	3.4E+10
Kr-88	2.2E+11
Kr-85m	2.1E+11
Xe-135	7.9E+11
Xe-133m	4.2E+09
Xe-133	2.2E+12
Xe-131m	7.3E+10
Kr-85	3.0E+11
I-131	1.8E+08
I-133	3.1E+08

[1] Hitachi-GE has calculated the discharges for two cases - CaseA: 12 month continued operation); and, Case B: 11 month continued operation + 1 month maintenance + one release from vacuum pump during start-up). These operations are assumed during normal operation in UK ABWR.) As the gaseous annual discharge of Case A is larger than ,Case B for all radionuclides (except for I-131) the discharges of Case A has been selected, except for I-131 where the value for Case B is selected due to the effect of the I-131 is released during 1 month maintenance.

5.3.2. Actual Measured values

There are a number of additional radionuclides in the European Commission (2004) document that have not been calculated in Japan. Where actual data exists for these radionuclides they have been used for the discharge rate. The available actual data are shown in Table 5.3-3.

Table 5.3-3: Annual discharge rate (actual Measured values)

Radionuclide	Annual discharge (Bq/y)	Reference source
H-3(gas)	8.0E+11	(12)
H-3(liquid)	6.4E+11	(12)
C-14(gas)	7.3E+11	(13)
Ar-41	5.9E+11	(14)

This measured data for H-3, C-14 and Ar-41 is considered appropriate to use for the UK ABWR assessment because the reference design is very similar to that used for units 6 and 7 at Kashiwazaki-Kariwa, from which the measured . Should any changes be made to the reference design during GDA, or beyond, the impacts of these changes on the measured data presented above will be assessed and adjusted as required.

5.3.3. Assumed values

It is considered that the possibility of the remaining radionuclides shown in the European Commission (2004) document arising and being discharged is quite low. Therefore, specific actual data is not available for them.

Requirements for the detection level of radioactive discharge monitoring equipment are prescribed in the Japanese Regulatory Guide for Radiation Monitoring of Effluent Released from Light Water Nuclear Power Reactor Facilities (15). There is no requirement in Japan to report discharges that are below these stipulated detection requirements, and such discharges are typically recorded as 'non-detectable' or 'ND'. It is possible to generate an estimate of the amount of radioactivity discharged for these radionuclides for which no other data is available by multiplying the detection limit of the discharge monitoring equipment by the total volume of gaseous waste discharged to the environment.

Therefore, the following assumption is made to evaluate the radioactivity discharged for these radionuclides, in both gaseous and liquid discharges:

$$\mathrm{Ni} = \mathrm{CDLi} \times \mathrm{Q}$$

Ni：radionuclide i annual discharge activity (Bq/y)

CDLi：radionuclide i requirement for the detection limit (Bq/cm^3) (15)

Q：annual discharge rate (cm^3/y)

The annual discharge rate of gaseous streams has been calculated from the summation of air flows through the HVAC systems associated with the Reactor Building, the Turbine Building and the Radioactive Waste Building. The following data have been obtained from design values associated with the standard ABWR:

- Reactor Building and Turbine Building: 498,000 m3/hr
- Radioactive Waste Building: 58,000 m3/hr
- Total ventilation system flow rate: 556,000 m^3/hr

This gives an equivalent annual ventilation system flow rate: 4,900,000,000 m^3/yr. The annual discharge rate of the liquid streams has been calculated from a total flow of 4700 m^3/yr across all relevant systems.

The resulting gaseous and liquid annual discharge rates for each radionuclide are shown in Table 5.3-4 and Table 5.3-5.

Table 5.3-4: Annual gaseous discharge rate (Assumed)

Radionuclide	Flow rate at HVAC (m^3/y)	Requirement for the Detection Limit (Bq/cm^3)	Annual Release(Bq/y)
I-132	4.9E+09	7.0E-08	3.4E+08
I-135		7.0E-08	3.4E+08
Sr-89		4.0E-10	2.0E+06
Sr-90		4.0E-10	2.0E+06
Zr-95		4.0E-09	2.0E+07
Nb-95		4.0E-09	2.0E+07
Cs-134		4.0E-09	2.0E+07
Cs-137		4.0E-09	2.0E+07
Ba-140		4.0E-09	2.0E+07
La-140		4.0E-09	2.0E+07
Ce-141		4.0E-09	2.0E+07
Ce-144		4.0E-09	2.0E+07
Pu-238		4.0E-10	2.0E+06
Pu-239+ Pu-240		4.0E-10	2.0E+06
Am-241		4.0E-10	2.0E+06
Cm-242		4.0E-10	2.0E+06
Cm-243		4.0E-10	2.0E+06
Cm-244		4.0E-10	2.0E+06
Cr-51		4.0E-09	2.0E+07
Mn-54		4.0E-09	2.0E+07
Co-58		4.0E-09	2.0E+07
Fe-59		4.0E-09	2.0E+07
Co-60		4.0E-09	2.0E+07
Zn-65		4.0E-09	2.0E+07
Ag-110m		4.0E-09	2.0E+07
Sb-122		4.0E-09	2.0E+07
Sb-124		4.0E-09	2.0E+07
Sb-125		4.0E-09	2.0E+07

Table 5.3-5: Annual liquid discharge rate (Assumed)

Radionuclide	Flow rate from HCW and LD (m^3/y)	Requirement for the Detection Limit (Bq/cm^3)	Annual Release(Bq/y)
Ru-103	4.7E+03	2.0E-02	9.4E+07
Ru-106		2.0E-02	9.4E+07
Te-123m		2.0E-02	9.4E+07
Zr-95		2.0E-02	9.4E+07
Nb-95		2.0E-02	9.4E+07
Ba-140		2.0E-02	9.4E+07
La-140		2.0E-02	9.4E+07
Ce-141		2.0E-02	9.4E+07
Ce-144		2.0E-02	9.4E+07
Pu-238		4.0E-03	1.9E+07
Pu-239 + Pu-240		4.0E-03	1.9E+07
Am-241		4.0E-03	1.9E+07
Cm-242		4.0E-03	1.9E+07
Cm-243		4.0E-03	1.9E+07
Cm-244		4.0E-03	1.9E+07
C-14		4.0E-02	1.9E+08
Fe-55		2.0E-02	9.4E+07
Ni-63		2.0E-02	9.4E+07
Zn-65		2.0E-02	9.4E+07
Ag-110m		2.0E-02	9.4E+07
Sb-122		2.0E-02	9.4E+07
Sb-124		2.0E-02	9.4E+07
Sb-125		2.0E-02	9.4E+07

5.4. Expected Events

Normal operations are defined within the Environment Agency's GDA P&ID as follows:

> *'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.*

Therefore, for Hitachi-GE to predict discharges and propose limits for the UK ABWR best performance should be estimated (as per the base load) and the activity released from any 'events that are expected to occur' or 'expected events' should also be added.

An expected event is an event whose frequency is relatively high compared to accident scenarios, but whose occurrence is unwanted. Measures are therefore put in place to manage the consequence of such events.

Using this definition, Hitachi-GE has undertaken a study to identify and address the events that are likely to occur and which would have a bearing on the discharges from the UK ABWR. The study has shown that the majority of perceived 'expected events' are indeed accidents or as a result of inadequate maintenance (i.e. a reduction in filter efficiency) and would be mitigated by adequate measures to prevent the situation from causing any impact (further information on this assessment will be provided in Step 2 GDA).

For example, the design of the UK ABWR ensures that all liquids generated from normal operations are collected into the liquid waste collection system, including those that are generated from small leaks and drips that occur during maintenance and changing of equipment. There is also no increase in radioactivity discharged as a result of degradation of equipment, because all treated liquid radioactive effluent is checked before discharge to ensure it complies with limits and to demonstrate that BAT has been applied. As the result, no increase of radioactivity in liquid discharges is expected as a consequence of these potential events.

The only expected event deemed to be valuable to be considered as part of limit setting was deemed to be leakage resulting from fuel pin failure. Should a fuel pin failure occur, it is detected and a process initiated to identify and isolate the fuel assembly in which the failed pin is located by systematically inserting control rods. Once the relevant assembly is isolated, the generation of fission products and their release into the coolant is significantly reduced, as detected at the inlet of the Off-Gas hold-up system. This isolation process, which can last up to 14 days (although typically 48 hours) has been estimated using the parameters below, and is included within the limit setting.

5.4.1. Noble Gas Release Ratio

Failure of a fuel pint increases the noble gas release ratio ('f-value'). Based on operational experience and expertise in Hitachi-GE shows that the best performance data (or 'realistic case') f-value is

f = 3.7E+7 Bq/s (30 min decay value). Should a fuel pin failure occur, Hitachi-GE's previous experience shows that a value of 3.7E+9 Bq/s is appropriate to calculate discharges.

5.4.2. Duration Time

When the pin failure occurs and the concentration of noble gas increases, the radiation monitor at the inlet of off-gas hold-up system detects the radioactivity increase (16). If the situation causing the increase in radioactivity continues, the operator can initiate the necessary measures to identify and isolate the fuel assembly in which the failed pin resides. After the pin failure fuel assembly is identified and isolated with control rods, the noble gas concentration will decrease and enter into a steady state i.e. the expected event has ended.

From operational experience it is estimated that the event duration is expected to range from 48 hours to 14 days depending on how quickly the fuel assembly is identified. For the purposes of estimating releases for GDA, 14 days is assumed as the duration time of the expected event. This is considered to be a cautious ‘worst case’ estimate. It is also not credible that the event would occur more than once in any fuel cycle. Table 5.4-1 shows the amount of radioactivity discharged from the stack to the environment during the expected event.

Table 5.4-1: Gaseous discharges for a fuel pin failure expected event

Radionuclide	Activity Discharge (Bq per event)
Kr-90	2.0E-04
Xe-139	2.3E-01
Kr-89	8.4E+09
Xe-137	3.1E+10
Xe-135m	5.5E+11
Xe-138	9.9E+11
Kr-87	6.7E+11
Kr-83m	1.3E+11
Kr-88	8.5E+11
Kr-85m	8.2E+11
Xe-135	3.0E+12
Xe-133m	1.6E+10
Xe-133	8.4E+12
Xe-131m	2.8E+11
Kr-85	1.1E+12
I-131	4.3E+08
I-133	1.2E+09

The ratio of gaseous discharges associated with the expected event to those expected for normal discharges (Table 5.3-2) is between 2.4 to 3.9 for all radionuclides. This ratio is likely to change as the anticipated discharges assessments are further refined. The values in Table 5.4-1 have been used in the prospective dose modelling assessment.

5.5. Short Term Release

In addition to the radiological dose implications of any expected events, Hitachi-GE is required to consider the potential dose implications of any short term releases from the UK ABWR.

The NDAWG-2-2011 (National Dose Assessment Working Group Guide) (17) defines a short term release as follows:

- *A release which is larger than a normal release (≥2% of 12-monthly actual or expected discharges) and occurs over a relatively short period of time (≤1 day). For a normally uniform discharge profile, this equates to about 1 week's discharge being released in 1 day or less;*

- *Releases that occur over longer periods of time (e.g. 5 days) may be considered as a continuous release, so long as the daily release during that period does not exceed 2% of the 12-month actual or expected discharges.*

Unlike the PWR, the ABWR does not discharge a large amount of radioactivity during routine start-up or shut-down operations. Therefore, the only significant short term release that needs consideration is that evident during the time of an expected fuel pin failure event. This may last up to 14 days, as discussed in the previous section. To evaluate its impact as a short term release, it is conservatively assumed that the total discharge associated with the expected event is discharged within 24 hours. Although this is not credible, it is anticipated that the release associated with the expected event will vary over the 14 days and so this approach provides a conservative estimate of the radiological consequences.

Table 5.4-1 shows the amount of radioactivity discharged from the stack to the environment during the short-term period of 24 hours (18). This data has also been used in the prospective dose modelling assessment (2).

6. Summary data

Table 6-1 presents the detailed nuclear reactions resulting in the production of activation products. Tables Table 6-2 and Table 6-3 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 production mechanism for a particular radionuclide, the route of that radionuclide to the environment, the treatment it may go through prior to its release to the environment and the quantification of the annual discharge from the approaches outlined in section 5.3 above. An indicative annual dose due to the discharge of each radionuclide is also included in the final column of each table.

The columns in the tables are:

- **Radionuclide**: The radionuclides listed are all those that may be released from an operation of a single unit UK ABWR.
- **Production mechanism**: A short description of the mechanism by which the radionuclide is produced, generating the activity e.g. X is a fission product from fuel, structural Uranium.
- **Route to the environment:** The route by which each radionuclide is released is described here and used as input to the prospective dose modelling (2).
- **Treatment:** Any treatment undertaken to abate or remove the radionuclide from the waste stream, or any change in its form.
- **Estimated annual discharge:** The total activity released during a rolling 12-month cycle.
- **Source of data:** An explanation of the derivation of the data for each radionuclide, with the approaches outlined in more detail in sections 5.3.1, 5.3.2 and 5.3.3.
- **Factor:** This is the multiplication factor that Hitachi-GE proposes for the consideration of limits for the UK ABWR at Step 1b. Selection of the multiplication factor is consistent with guidance previously issued by the Environment Agency (18).

  The three techniques used in this report to estimate discharges introduce varying degrees of caution in establishing the headroom that should apply to each of the proposed limits. For Step 1b a multiplication factor of 2 has been selected to allow for the inherent uncertainty with any new build design project and to allow room for minor design modifications and decisions to be made without the need to revisit the limit setting.

Further assessment of the source term and associated discharges will be performed at Step 2 and, if appropriate, this will allow the headroom for the UK ABWR limits to be refined to the extent that a multiplication factor may no longer be necessary.

- **Proposed annual discharge limit:** This will be 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 x 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 potential dose received from operating a single UK ABWR at the proposed limits for 12 months. The values in Tables Table 6-2 and Table 6-3 have been calculated and reported in the Prospective Dose Modelling report (2) and are based on Stage 2 of the Environmental Agency's initial radiological assessment methodology.
- **Significant (Y/N):** Is the radionuclide considered 'Significant' (as per the methodology outlined in Section 7) and therefore selected for consideration as a permitted radionuclide.

Table 6-1: Detailed Nuclear Reactions for Activation Products

Radioisotope	Nuclear production mechanisms
H-3	B10 (n,2a) H3 (from Boron in control rods) Be9 (n, Li7) H3 Be9 (n,He6) H3 H2 (n,g) H3 (from H-2 in reactor water -negligible)
C-14	N14 (n,p) C14 O17 (n,a) C14 C13 (n,g) C14 (from structural materials)
Ar-41	Ar40 (n,g) Ar41(activation of atmospheric Ar in coolant)
Cr-51	Cr50 (n,g) Cr51
Mn-54	Fe54 (n,p) Mn54 Mn55 (n,2a) Mn54
Fe-55	Fe54(n,g)Fe55 Fe56(n,2n)Fe55
Co-58	Ni58 (n,p) Co58
Fe-59	Fe58 (n,g) Fe59
Co-60	Co59 (n,g) Co60
Ni-63	Ni62 (n,g) Ni63
Zn-65	Zn64 (n,g) Zn65
Ag-110m	Ag109 (n,g) Ag110m

Table 6-2: Summary table for Gaseous releases (1/2)

Radionuclide	Production mechanism	Route to the environment	Treatment	Annual Discharge (Bq/y)	Data source (C/M/A)	Factor	Expected Event Discharges (Bq)	Proposed Discharge Limit (Bq/y)	Dose at Limits (μSv/y)	Significant (Y/N)
Ar-41	Ar40(n,g)Ar41 Activation of entrained atmospheric Ar in coolant	Activation of coolant Migration into steam Separation at condenser Discharge via stack Adventitious discharges from steam leaks Discharge via HVAC	Off-gas treatment system carbon delay beds Discharge at height via main stack	5.9E+11	M	2	-	1.2E+12	1.73E-01	Y
Kr-85	Fission product from fuel, structural Uranium	Migration into reactor water (direct or through pin failure) 100 % migration into steam Separation at condenser Discharge via stack via off-gas system	Off-gas treatment system carbon delay beds Discharge at height via main stack	3.0E+11	C	2	1.10E+12	1.7E+12	9.73E-04	Y
Kr-85m				2.1E+11	C		8.20E+11	1.2E+12	1.97E-02	
Kr-87				1.8E+11	C		6.70E+11	1.0E+12	9.46E-02	
Kr-88				2.2E+11	C		8.50E+11	1.3E+12	3.01E-01	
Kr-89				2.2E+09	C		8.40E+09	1.3E+10	3.01E-03	
Xe-131m				7.3E+10	C		2.80E+11	4.3E+11	3.58E-04	
Xe-133				2.2E+12	C		8.40E+12	1.3E+13	4.10E-02	
Xe-133m				4.2E+09	C		1.60E+10	2.4E+10	7.32E-05	
Xe-135				7.9E+11	C		3.00E+12	4.6E+12	1.21E-01	
Xe-135m				1.4E+11	C		5.50E+11	8.3E+11	3.76E-02	
Xe-137				8.1E+09	C		3.10E+10	4.7E+10	1.09E-02	
Xe-138				2.6E+11	C		9.90E+11	1.5E+12	1.96E-01	
I-131	Fission product from fuel, structural Uranium	Migration into reactor water (direct or through pin fracture) Partial migration into steam Separation at condenser (g.) Discharge via stack via off-gas system (g.) Discharge via HVAC of volatile Iodine in aqueous stream	Off-gas treatment system Carbon delay beds Discharge at height via main stack	1.8E+08	C	2	4.30E+08	7.9E+08	1.08E+00	Y
I-132				3.4E+08	A		-	6.8E+08	4.62E-04	N
I-133				3.1E+08	C		1.2E+09	1.8E+09	5.12E-02	N
I-135				3.4E+08	A		-	6.8E+08	1.28E-03	N
Sr-89	Fission product from fuel, structural Uranium	Migration into reactor water (direct or through pin failure) Entrainment of aerosol into steam lines Discharge via condenser and stack	Coarse filters to protect Charcoal Beds Removal by Charcoal Beds Discharge at height via main stack	2.0E+06	A	2	-	4.0E+06	5.24E-05	Y
Sr-90				2.0E+06	A		-	4.0E+06	9.89E-04	Y
Zr-95				2.0E+07	A		-	4.0E+07	8.80E-04	N
Nb-95				2.0E+07	A		-	4.0E+07	2.68E-04	N
Cs-134				2.0E+07	A		-	4.0E+07	1.30E-02	N
Cs-137				2.0E+07	A		-	4.0E+07	1.69E-02	N
Ba-140				2.0E+07	A		-	4.0E+07	4.51E-04	N
La-140				2.0E+07	A		-	4.0E+07	8.72E-05	N
Ce-141				2.0E+07	A		-	4.0E+07	1.56E-04	N
Ce-144				2.0E+07	A		-	4.0E+07	1.89E-03	N

[1] C = Calculated value (see section 5.3.1); M = Actual Measured Data (see section 5.3.2); A = Assumed Value (see section 5.3.3)

Table 6-2: Summary table for Gaseous releases (2/2)

Radionuclide	Production mechanism	Route to the environment	Treatment	Annual Discharge (Bq/y)	Data source (C/M/A)[1]	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	Coarse filters to protect Charcoal Beds Removal by Charcoal Beds Discharge at height via main stack	2.0E+06	A	2	-	4.0E+06	1.81E-01	N
Pu-239 + Pu240	Activation of fuel U by neutron capture			2.0E+06	A		-	4.0E+06	1.99E-01*	N
Am-241	Activation of fuel U by neutron capture			2.0E+06	A		-	4.0E+06	1.70E-01	N
Cm-242	Activation of fuel U by neutron capture			2.0E+06	A		-	4.0E+06	2.16E-02	N
Cm-243	Activation of fuel U by neutron capture			2.0E+06	A		-	4.0E+06	1.27E-01	N
Cm-244	Activation of fuel U by neutron capture			2.0E+06	A		-	4.0E+06	1.10E-01	N
Total-alpha	Activation of fuel U by neutron capture			N/A	N/A		-			Y
H-3	Ternary fission in fuel B10 (n,2a) H3 from Boron in control rods Be9 (n,x) H3 from secondary neutron sources H2 (n,g) H3 from H-2 in reactor water (negligible)	Migration into reactor water (direct or through pin failure or diffusion through pin cladding) Entrainment of aerosol into steam lines Adventitious discharges from steam leaks Discharge via HVAC of aqueous vapour Discharge via condenser and stack	None	8.0E+11	M	2	-	1.6E+12	1.92E-01	Y
C-14	N14 (n,p) C14 O17 (n,a) C14 both from fuel and reactor water C13 (n,g) C14 from structural materials	Depends on water chemistry Partition liquid / vapour of CO_2 Discharge via HVAC	None	7.3E+11	M	2	-	1.5E+12	1.87E+01	Y
Cr-51	Cr50 (n,g) Cr51 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 Aerosol generation in tanks, pools and release via HVAC	Coarse filters to protect Charcoal Beds Removal by Charcoal Beds Discharge at height via main stack Filtration on HVAC discharge Discharge at height via main stack	2.0E+07	A	2	-	4.0E+07	9.51E-06	N
Mn-54	Fe54 (n,p) Mn54 Mn55 (n,2a) Mn54 Activation product			2.0E+07	A		-	4.0E+07	2.07E-03	N
Co-58	Ni58 (n,p) Co58 Activation product			2.0E+07	A		-	4.0E+07	6.09E-04	N
Fe-59	Fe58 (n,g) Fe59 Activation product			2.0E+07	A		-	4.0E+07	7.21E-04	N
Co-60	Co59 (n,g) Co60 Activation of reactor components Activation of insoluble and soluble metal crud and particulate in reactor water			2.0E+07	A		-	4.0E+07	2.09E-02	N
Zn-65	Zn64 (n,g) Zn65 Activation product			2.0E+07	A		-	4.0E+07	1.36E-02	N
Ag-110m	Ag109 (n,g) Ag110m Activation product			2.0E+07	A		-	4.0E+07	8.72E-03	N
Sb-122	Activation of Sb present in structural components such as bearings			2.0E+07	A		-	4.0E+07	1.69E-02†	N
Sb-124				2.0E+07	A		-	4.0E+07	1.69E-02†	N
Sb-125				2.0E+07	A		-	4.0E+07	2.61E-03	N
	*based on Pu239 †based on Cs137							total	2.2E+01	

Table 6-3: Summary table for Liquid releases (1/2)

Radionuclide	Production mechanism	Route to the environment	Treatment	Annual Discharge (Bq/y)	Data source (C/M/A)[1]	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) Build-up in reactor and fuel pool water Spillages gathered in sumps Discharge via CAD, LD sump Discharge via D/W HCW Sump - occasional	CUW system LD prefilter LD AC filter LD filter D/W HCW Evaporator D/W HCW IEX	9.4E+07	A	2	1.9E+08	1.29E-04	N
Ru-106				9.4E+07	A		1.9E+08	7.02E-04	N
Te-123m				9.4E+07	A		1.9E+08	2.19E-03	N
Sr-89				1.6E+06	C		3.2E+06	3.70E-07	Y
Sr-90				8.2E+05	C		1.6E+06	7.52E-07	Y
Zr-95				9.4E+07	A		1.9E+08	1.27E-03	N
Nb-95				9.4E+07	A		1.9E+08	3.22E-04	N
I-131		Migration into reactor water (direct or through pin failure) Partial migration into steam Build-up in reactor and fuel pool water Spillages gathered in sumps Discharge via CAD, LD sump Discharge via D/W HCW Sump - occasional		1.6E+06	C	2	3.2E+06	6.16E-07	Y
Cs-134	Fission product from fuel, structural Uranium	Migration into reactor water (direct or through pin failure) Build-up in reactor and fuel pool water Spillages gathered in sumps Discharge via CAD, LD sump Discharge via D/W HCW Sump - occasional	CUW system LD prefilter LD AC filter LD filter D/W HCW Evaporator D/W HCW IEX	4.1E+06	C	2	8.2E+06	7.58E-05	N
Cs-137				6.5E+06	C		1.3E+07	1.50E-04	Y
Ba-140				9.4E+07	A		1.9E+08	7.32E-05	N
La-140				9.4E+07	A		1.9E+08	2.19E-05	N
Ce-141				9.4E+07	A		1.9E+08	2.49E-05	N
Ce-144				9.4E+07	A		1.9E+08	2.19E-04	N
Pu-238	Activation of fuel U by neutron capture	Migration into reactor water (direct or through pin failure) Build-up in reactor and fuel pool water Spillages gathered in sumps Discharge via CAD, LD sump Discharge via D/W HCW Sump - occasional	CUW system LD prefilter LD AC filter LD filter D/W HCW Evaporator D/W HCW IEX	1.9E+07	A	2	3.8E+07	4.68E-03	N
Pu-239 + Pu-240				1.9E+07	A		3.8E+07	4.97E-03	N
Am-241				1.9E+07	A		3.8E+07	2.08E-04	N
Cm-242				1.9E+07	A		3.8E+07	8.49E-06	N
Cm-243				1.9E+07	A		3.8E+07	8.78E-04	N
Cm-244				1.9E+07	A		3.8E+07	7.90E-05	N
Total-alpha				N/A	N/A				Y

[1] C = Calculated value (see section 5.3.1); M = Actual Measured Data (see section 5.3.2); A = Assumed Value (see section 5.3.3)

Table 6-3: Summary table for Liquid releases (2/2)

Radionuclide	Production mechanism	Route to the environment	Treatment	Annual Discharge (Bq/y)	Data source (C/M/A)[2]	Factor	Proposed Discharge Limit (Bq/y)	Dose at Limits (μSv/y)	Significant (Y/N)
H-3	Ternary fission in fuel B10 (n,2a) H3 (from Boron in control rods) Be9 (n, Li7) H3 Be9 (n,He6) H3(from secondary neutron sources) H2 (n,g) H3 (from H-2 in reactor water -negligible)	Migration into reactor water (direct or through pin failure) Build-up in reactor and fuel pool water Spillages gathered in sumps Discharge via CAD, LD sump Discharge via D/W HCW Sump - occasional	CUW system LD prefilter LD AC filter LD filter D/W HCW Evaporator D/W HCW IEX	6.4E+11	M	2	1.3E+12	8.91E-05	Y
C-14	N14 (n,p) C14 (fuel and reactor water) O17 (n,a) C14 (fuel and reactor water) C13 (n,g) C14 from structural materials	Migration into reactor water (direct or through pin failure) Build-up in reactor and fuel pool water Spillages gathered in sumps Discharge via CAD, LD sump Discharge via D/W HCW Sump - occasional		1.9E+08	A	2	3.8E+08	1.35E-02	Y
Cr-51	Cr50 (n,g) Cr51 Activation of reactor components, insoluble and soluble metal crud and particulate in reactor water	Build-up in reactor and fuel pool water Spillages gathered in sumps Discharge via CAD, LD sump Discharge via D/W HCW Sump - occasional	CUW system LD prefilter LD AC filter LD filter D/W HCW Evaporator D/W HCW IEX	1.6E+06	C	2	3.2E+06	1.48E-07	N
Mn-54	Fe54 (n,p) Mn54 Mn55 (n,2a) Mn54 Activation product			3.3E+07	C	2	6.6E+07	1.17E-03	N
Fe-55	Fe54(n,g)Fe55 Fe56(n,2n)Fe55			9.4E+07	A	2	1.9E+08	4.39E-06	N
Co-58	Ni58 (n,p) Co58 Activation product			2.5E+06	C	2	5.0E+06	2.66E-05	N
Fe-59	Fe58 (n,g) Fe59 Activation product			5.7E+06	C	2	1.1E+07	4.15E-05	N
Co-60	Co59 (n,g) Co60 Activation of reactor components, insoluble and soluble metal crud and particulate in reactor water			2.5E+07	C	2	5.0E+07	1.08E-02	N
Ni-63	Ni62 (n,g) Ni63			9.4E+07	A	2	1.9E+08	5.27E-05	N
Zn-65	Zn64 (n,g) Zn65 Activation product			9.4E+07	A	2	1.9E+08	4.97E-02	N
Ag-110m	Ag109 (n,g) Ag110m Activation product			9.4E+07	A	2	1.9E+08	5.85E-02	N
Sb-122	Activation of Sb present in structural components such as bearings			9.4E+07	A	2	1.9E+08	2.19E-03	N
Sb-124				9.4E+07	A	2	1.9E+08	2.19E-03	N
Sb-125				9.4E+07	A	2	1.9E+08	4.24E-04	N
							Total	1.55E-01	

7. Selection of Significant Radionuclides

7.1. Selection process

The basis upon which radionuclides are selected to be permitted for discharge by the Environment Agency is published in Environment Agency guidance (9). As stated in the guidance, the Environment Agency will normally set annual site limits for each radionuclide, or group of radionuclide(s) that meet certain parameters during for normal operation. The parameters are given as radionuclides or groups of radionuclides that:

a. <u>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);</u>

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 6-3, there are no radionuclides that are in the liquid discharge that fall into this category. However, as presented in Table 6-2, C-14 and I-131 discharged via the gaseous discharge route are predicted to have indicative doses in excess of 1 µSv/y.

b. <u>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/hour);</u>

There are no radionuclides that fall into this category, as demonstrated by the prospective dose modelling that Hitachi-GE has undertaken (2). 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. <u>are significant in terms of the quantity of radioactivity discharged (that is, the discharge of a radionuclide exceeds 1 TBq per year);</u>

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 presented in section 6.2 and summarised in Tables 7.2-1 and 7.2-2). As can be seen, the data shows that there are a series of individual radionuclides, as well as a single group of radionuclides, that, 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 1.6 TBq, if discharged at the limit for 12 months;
- Gaseous C-14 with a predicted annual discharge of 1.5 TBq , if discharged at the limit for 12 months;
- Gaseous Ar-41 with a predicted annual discharge of 1.2 TBq, if discharged at the limit for 12 months;
- Noble Gases (group) with a predicted annual discharge of 26.8 TBq (including Ar-41), if discharged at the limit for 12 months; and,
- Liquid Gaseous H-3 with a predicted annual discharge of 1.3 TBq, if discharged at the limit for 12 months.

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 (2), 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;

It is not thought that any additional radionuclides will be included within the selection of significant radionuclides as a result of this requirement. This will be reviewed in more detail during Step 2.

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

As outlined in the Demonstration of BAT report (5), the following radionuclides are considered good indicators of plant performance and are comparable to the series of radionuclides sampled and monitored in the ABWR units that are operating in Japan.

For gaseous releases:

- Iodine-131
- Iodine-133
- Gamma emitters such as Chromium-51, Manganese-54, Iron-59, Cobalt-60, Caesium-134, Caesium-137
- Strontium-89, Strontium-90
- Total Beta
- Total Alpha

For liquid releases:

- Gamma emitters such as Chromium-51, Manganese-54, Iron -59, Cobalt-58, Cobalt-60, Iodine-131, Caesium-134, Caesium-137
- Strontium-89, Strontium-90
- Total Beta
- Total Alpha

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.

It is not thought that any additional radionuclides will be included within the selection of significant radionuclides as a result of this requirement. This will be reviewed in more detail during Step 2.

7.2. Proposal of UK ABWR limits for Step 1b

Drawing on the relevant information in Tables 6-2 and 6-3, Table 7.2-1 and Table 7.2-2 summarise the limits Hitachi-GE propose for the UK ABWR:

Table 7.2-1: Discharges to Liquid

Radionuclide (or group)	Limit – 12 month rolling	Suggested rounded limit
Tritium	1.30E+12	1.3E+12
Carbon-14	3.80E+08	3.8E+08
Caesium-137	1.30E+07	1.3E+07
Strontium (Sr89 and Sr90)	4.80E+06	4.8E+06
Iodine – 131	3.20E+06	3.2E+06
Other Beta/Gamma	3.18E+09	3.2E+09
Total Alpha	2.28E+08	2.3E+08

Table 7.2-2: Discharges to Air

Radionuclide (or group)	Limit – 12 month rolling	Suggested rounded limit
Tritium	1.60E+12	1.6E+12
Carbon-14	1.50E+12	1.5E+12
Argon-41	1.20E+12	1.2E+12
Noble Gases (group, incl. Ar-41)	2.68E+13	2.7E+13
Iodine – 131	7.90E+08	7.9E+08
Iodines (group, incl. I-131)	3.95E+09	4.0E+09
Strontium (Sr-89 and Sr-90)	8.00E+06	8.0E+06
Other Beta/Gamma	7.20E+08	7.2E+08
Total Alpha	2.40E+07	2.4E+07

8. Conclusion

The joint regulatory guidance ‘Principles for the Assessment of Prospective Public Doses arising from Authorised Discharges of Radioactive Waste to the Environment’ (19) describes the various dose limits and dose constraints that apply to prospective dose assessments for members of the public. This information is reproduced here as Table 8-1 for comparison against the data presented in Table 6-2 and 6-3, 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 exposure 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 22 μSv y^{-1} and 0.15 μSv y^{-1}, respectively. These estimates are based on the discharges presented in Tables 6-2 and 6-3 and a dose assessment based on the Environment Agency IRA stage 2 methodologies. Further information can be found in the Prospective Dose Modelling Report (2).

These estimates are well below the legal limit of 1mSv per year and the source constraint of 0.3 mSv per year. It should however be noted that these indicative prospective annual doses are from discharges only and do not include the contribution from direct radiation. For a proper comparison to be undertaken, the prospective dose and the direct radiation should be considered in totality; however, the source dose constraint is significantly higher than the predicted dose and so it is expected that the sum of the prospective dose from discharges and the direct radiation will still be within the source constraint of 0.3 mSv per year.

In conclusion, it is considered likely that the prospective dose assessment and the direct radiation due to the normal operation of a single ABWR unit will fall between the threshold of optimisation and the source constraint.

Table 8-1: Dose limits/Constraints

Effective dose criteria	**Dose quantity**	**Application to prospective dose assessments**	**Purpose of assessment**
Dose limit	1 mSv/y	One or more future discharges is 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 threshold of optimisation 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. Doses sufficiently low to be used as an input to the optimisation process without further refinement