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UK ABWR

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UK ABWR Generic Design Assessment

Demonstration of BAT

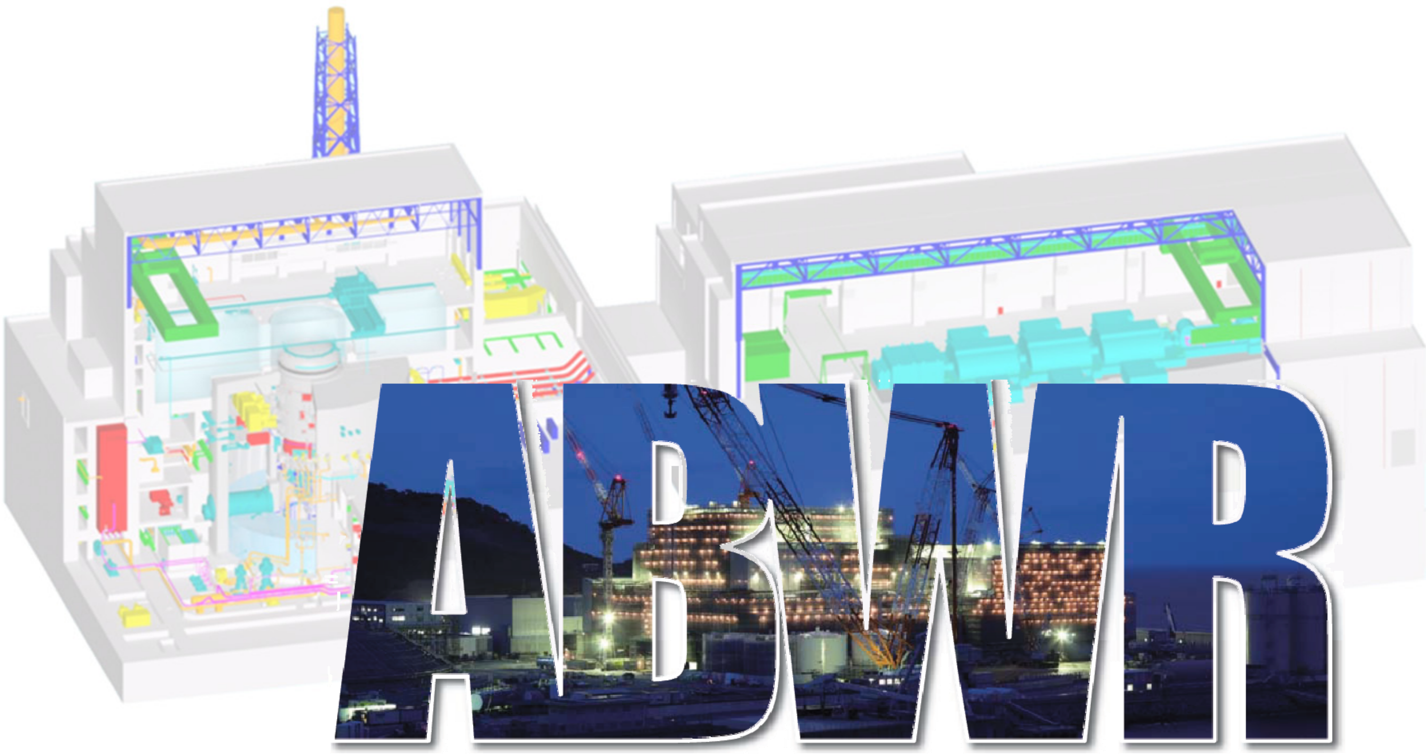

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Table of Contents

1. Acronyms

ABWR	Advanced Boiling Water Reactor
ALARA	As Low As Reasonably Achievable
ASTM	American Society for Testing and Materials
BAT	Best Available Technique
BSC	Basis of Safety Case
Bq	Becquerel
BSW	Biological Shield Wall
BWR	Boiling Water Reactor
C	Copyright
CAD	Controlled Area Drain System
CD	Condensate Demineraliser
CEM	Continuous Emission Monitor
CF	Condensate Filter
CP	Corrosion Product
CILC	Crud-Induced Localised Corrosion
COTS	Commercial Off-The-Shelf
CONW	Concentrated Waste System
CST	Condensate Storage Tank
CUW	Reactor Water Clean-up System
D/S	Dryer/Separator
D/W	Dry Well
DEFRA	Department for Environment, Food and Rural Affairs
DF	Decontamination Factor
DP	Differential Pressure
DPUR	Dose Per Unit Release
DZO	Depleted Zinc Oxide
EARWG	Environment Agencies Requirements Working Group
ECP	Electrochemical Corrosion Potential
ENUSA	Enusa Industrias Avanzandas
EPR	European Pressurised Water Reactor
FBS	Fuel Business System
FA	Forward Action
FAC	Flow Accelerated Corrosion
FD	Filter Demineraliser
FHM	Fuel Handling Machine
FP	Fission Product
FPC	Fuel Pool Cooling and Clean-up System

FWP	Feed Water Pump
GDA	Generic Design Assessment
GDF	Geological Disposal Facility
GENSUA	GNF ENUSA Nuclear Fuel S.A.
GNF	Global Nuclear Fuel
GSC	Gland Steam Condenser
GWd	Gigawatt-day
GWd/t	Gigawatt-day/tonne
HAW	Higher Activity Waste
HCW	High Chemical Impurities Waste
HEPA	High Efficiency Particulate Air Filter
HFF	Hollow Fibre Filter
HLW	High Level Waste
HOP	Hydrazine, Oxalic Acid, Potassium Permanganate
HPCP	High Pressure Condensate Pump
HPHD	High Pressure Heater Drain
HVAC	Heating Ventilating and Air Conditioning
HWC	Hydrogen Water Chemistry
IAEA	International Atomic Energy Agency
IGSCC	Intergranular Stress Corrosion Cracking
ILW	Intermediate Level Waste
IRAT	Initial Radiological Assessment Tool
ISO	International Standards Organisation
LCW	Low Chemical impurities Waste
LCM	Low Cobalt Material
LD	Laundry Drain
LDS	Leak Detection System
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LoC	Letter of Compliance
LPHD	Low Pressure Heater Drain
LPRM	Local Power Range Monitor
LTP	Lower Tie Plate
LWR	Light Water Reactor
LWMS	Liquid Waste Management System
MCR	Main Control Room
MIDAS	Multi-Inspection and Data Acquisition System
MPS	Missing Pellet Surface
MSIV	Main Steam Isolation Valves
MUWC	Make Up Water Condensate System

MVP	Mechanical Vacuum Pump
NDA	Nuclear Decommissioning Authority
NMCA	Noble Metal Chemical Addition
NPP	Nuclear Power Plant
NWC	Normal Water Chemistry
NZO	Natural Zinc Oxide
OECD	Organisation for Economic Co-Operation and Development
OD	Outer Diameter
OG	Off-Gas System
OLNC	On-Line Nobel Chem™
ONR	Office for Nuclear Regulation
OSPAR	Oslo and Paris Convention on Protection of the Marine Environment of the North East Atlantic
P&ID	Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs
PCI	Pellet Cladding Interaction
PCSR	Pre Construction Safety Report
PCV	Primary Containment Vessel
PLR	Primary Loop Recirculation System
PPE	Personal Protective Equipment
ppb	Parts Per Billion
ppm	Parts Per Million
PST	Power Suppression Testing
PWR	Pressurised Water Reactor
QA	Quality Assurance
R/B	Reactor Building
RCA	Radiological Controlled Area
RCCV	Reinforced Concrete Containment Vessel
RCW	Reactor Building Cooling Water System
REP	Radioactive Substances Regulation – Environmental Principle
RGP	Relevant Good Practice
RHR	Residual Heat Removal System
RIN	Reactor Internals
RIP	Reactor Internal Pump
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RSR	Radioactive Substances Regulation
RSW	Reactor Shield Wall
RWM	Radioactive Waste Management Limited
RW/B	Radwaste Building

SAP	Safety Assessment Principles
S/B	Service Building
S/C	Suppression Chamber
SCC	Stress Corrosion Cracking
SCV	Steel Containment Vessel
SF	Spent Fuel
SFP	Spent Fuel Storage Pool
SGTS	Standby Gas Treatment System
SHE	Standard Hydrogen Electrode
SJAE	Steam Jet Air Ejector
S/P	Suppression Pool
SPCU	Suppression Pool Clean-up System
SRV	Safety Relief Valve
SSR	Steam Seal Regulator
SS	Spent Resin & Sludge
SSSS	Separate steam seal system
Sv	Sievert
SWF	Solid Waste Facility
SWMS	Solid Waste Management System
T/B	Turbine Building
T/D-RFP	Turbine Driven Reactor Feedwater Pump
TG	Turbine Gland
TGS	Turbine Gland Steam System
TGSCC	Trans Granular Stress Corrosion Cracking
TMOL	Thermal-Mechanical Operating Limits
TOC	Total Organic Carbon
TVTS	Tank Vent Treatment System
UK	United Kingdom
U.S.	United States of America
UT	Ultrasonic Testing
VLLW	Very Low Level Waste
WAC	Waste Acceptance Criteria
WILW	Wet-Solid Intermediate Level Waste
WLLW	Wet-Solid Low Level Waste
WSC	Waste Service Contract
WPM	Wear Proof Material

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3. Introduction

This report presents the Claims, Arguments, Evidence that have been developed to demonstrate the application of Best Available Techniques (BAT) to the United Kingdom (UK) Advanced Boiling Water Reactor (ABWR). In doing so, this report demonstrates that the environmental performance associated with the practice of generating electricity from the UK ABWR will be optimised, and that impacts from potentially harmful ionising radiation on members of the public and the environment are being minimised.

This report forms part of Hitachi-GE's Generic Design Assessment (GDA) submission for the Environment Agency's assessment. The Environment Agency's requirements for undertaking environmental optimisation of discharges of radioactivity and to demonstrate the application of BAT for the GDA submission are defined within the Environment Agency's Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs (P&ID) [Ref-1].

These requirements are consistent with the Environment Agency's standard Environmental Permit conditions [Ref-2] that would apply to any future operator of a UK ABWR. To ensure compatibility of the generic environmental permit application with all future site specific permit applications Hitachi-GE has elected to use these standard BAT conditions. These conditions have been used to form the Claims that are a key part of the approach adopted to demonstrate the application of BAT. This approach is described in full in the 'Approach to Optimisation' report [Ref-3] and uses the Claim, Argument, Evidence approach. These two reports should be read in conjunction to provide a complete understanding of Hitachi-GE's approach and understanding of the derivation of the Claims and Arguments made in this report and the Evidence that has been gathered to substantiate them.

4. Approach to Environmental Optimisation and the Application of Best Available Techniques

4.1. Regulatory Environmental Principles

In its 'Approach to optimisation' report [Ref-3] Hitachi-GE described a methodology that sets out the approach to environmental optimisation and the application of BAT to the UK ABWR. This methodology is considered to be consistent with industry Relevant Good Practice (RGP) and takes into account the relevant Radioactive Substances Regulation – Environmental Principles (REP) [Ref-4] and the principles of optimisation set out in the Radioactive Substances Regulation (RSR): Principles of optimisation in the management and disposal of radioactive waste [Ref-5]. Hitachi-GE's 'Consideration of and compliance with the REPs' report [Ref-6] details the approach undertaken by Hitachi-GE to reviewing and incorporating each of the relevant REPs within the GDA submission and highlights the REPs specifically taken into account in each report. Within this report, each of the Claims highlights the key REPs that it addresses. A summary of the regulatory context that defines the requirement for environmental optimisation and the application of BAT is provided within the 'Approach to Optimisation' report [Ref-3].

4.2. Hitachi-GE BAT Philosophy

In the case of demonstrating BAT, the objective of Hitachi-GE's approach to environmental optimisation is to deliver the following:

- Protection of members of the public from exposure to potentially harmful ionising radiation and reduce any doses to As Low As Reasonably Achievable (ALARA);
- protecting the environment within which we operate and live;
- enabling the Nuclear Power Plant (NPP) to operate efficiently;
- enhance reputation as a 'good neighbour'; and
- complying with regulations (as summarised in Hitachi-GE's 'Approach to Optimisation' [Ref-3]).

Hitachi-GE's approach is guided by the following principles:

- **Evolution of the UK ABWR design.** BWRs benefit from a long operational history, which has enabled operational feedback to inform the design. Safety, environment and operability have all influenced how the design has evolved at each design iteration. Through the application of this BAT methodology Hitachi-GE will demonstrate how the design has evolved, minimising discharges to the environment and doses to the public.

- **Integration of the BAT methodology into decision making**. There are several considerations that need to be borne in mind when making decisions on the design and future operation of a NPP. Some of these are directly attributed to the Office for Nuclear Regulation (ONR) requirements, for example, the reduction of worker dose to ALARP, whereas others are less specific, such as "trouble" (e.g. ease of implementation, operability and decommissioning implications) or "technology maturity". Importantly – the demonstration of BAT needs to be integrated into the project programme and decision making process.
- **Opportunity**. Recognising that the demonstration of BAT should cover the lifecycle of the plant, certain elements will be best addressed during GDA whereas others will be better managed at a site specific level. In conjunction with future operators, Hitachi-GE has endeavoured to identify the best time to deliver elements of the programme to ensure that opportunities to further optimise the UK ABWR can be realised.

The key steps of the BAT methodology adopted by Hitachi-GE are summarised within Figure 4.2-1 below. For a full description, see reference [Ref-3].

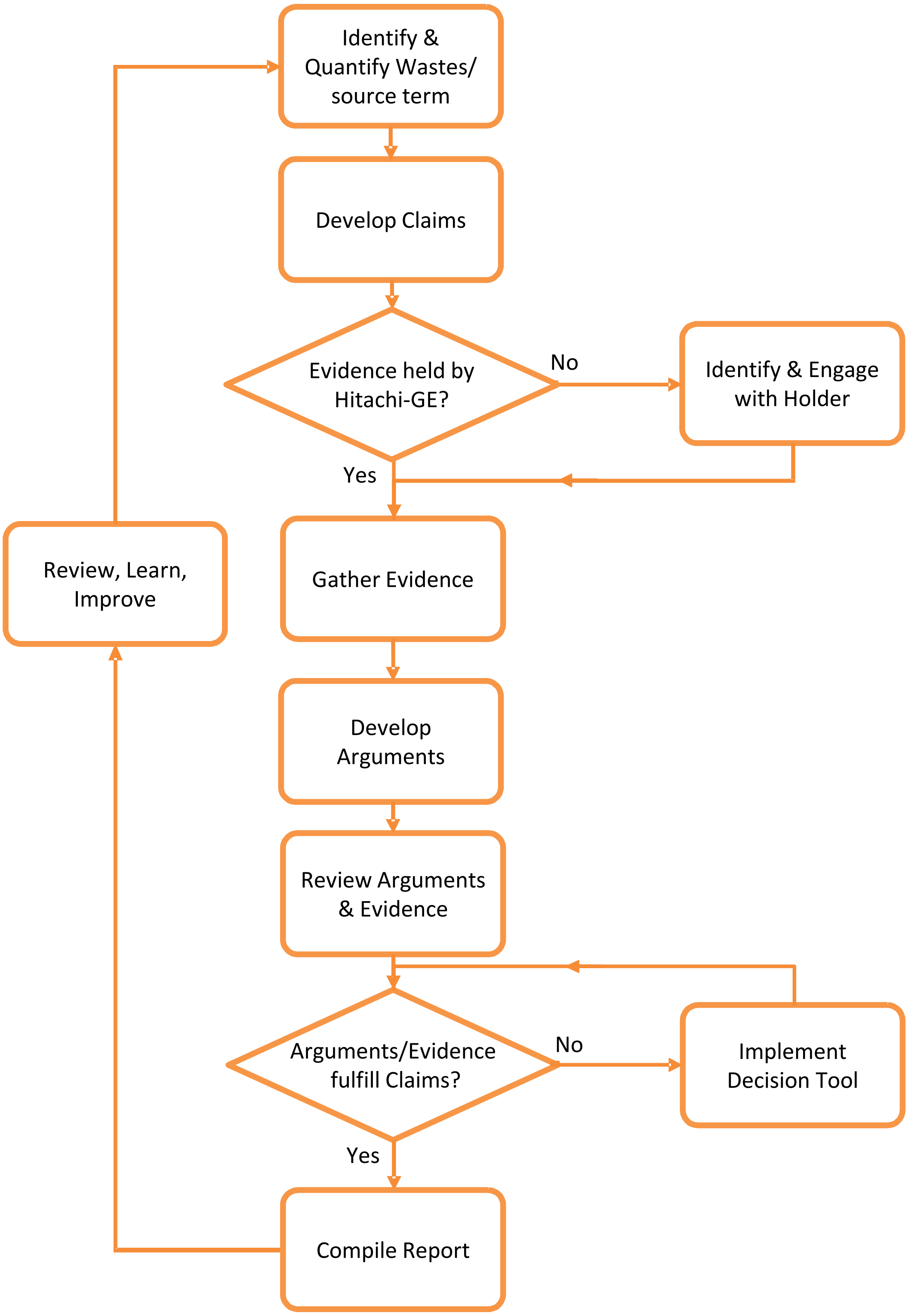


Figure 4.2-1: Methodology for Demonstrating the Application of BAT

4.3. Context of Optimisation within the GDA Process

The demonstration of environmental optimisation and the application of BAT for the UK ABWR will take place across all stages of the project life cycle (from the design stage through to decommissioning). The approach to environmental optimisation adopted by Hitachi-GE as set out in the 'Approach to Optimisation' [Ref-3] provides a framework that supports both the evolution that takes place as a design develops and the iterative nature of any BAT demonstration. This ensures that opportunities are both explored and realised during that phase of a project's life cycle where they are expected to deliver the greatest benefit. The environmental optimisation of the UK ABWR is considered to be focused on two key themes:

- **Engineered systems and controls**. The development of engineering systems and controls that contribute to the prevention and minimisation of radioactive waste generation and discharges and that minimise the impact of any discharges on members of the public and the environment.
- **Management systems and controls**. The operation of the engineered systems and controls will be further optimised through the development and implementation of management systems and controls.

The demonstration of environmental optimisation has been undertaken in parallel to the demonstration that the design and operation of the UK ABWR reduces health and safety risks to ALARP. This optimisation process is ongoing and as such has the potential to result in changes being made to the plant design. If changes are required they will be subject to the Hitachi-GE change management process which includes the requirement to apply and to demonstrate the application of BAT.

At the GDA stage it is considered important to demonstrate the application of BAT in relation to those features of the design that could foreclose options at later stages of the project's life cycle. The GDA demonstration of BAT will therefore focus on engineered systems and controls. Where the design will be influenced by site specific factors, consideration on how to prevent the foreclosure of options by providing flexibility within the design will be set out within the supporting Evidence. Whilst it is also recognised that the management system and controls will be developed during the site specific assessment the following will be considered during the Evidence gathering stage of the GDA:

- How engineered systems and controls contribute to the effectiveness of management systems and controls and vice versa.
- Human factors (HF) relating to management systems and controls to ensure that they can be effectively implemented.

HF considerations relating to both management systems and engineered controls have and will continue to be incorporated into the design of the UK ABWR under the HF Integration Plan. Where HF are relied upon as part of the BAT case, these instances will be fed into the Human Factors Integration (HFI) process outlined in PCSR Chapter 27 Human Factors [Ref-99].

4.4. Application of the Decision Tool

An inherent part of the approach adopted by Hitachi-GE to both apply BAT (implementation within the design of the UK ABWR) and in demonstrating the application of BAT (production of this 'Demonstration of BAT' report) is the decision tool process set out in the 'Approach to Optimisation' [Ref-3] and reproduced as Figure 4.4-1.

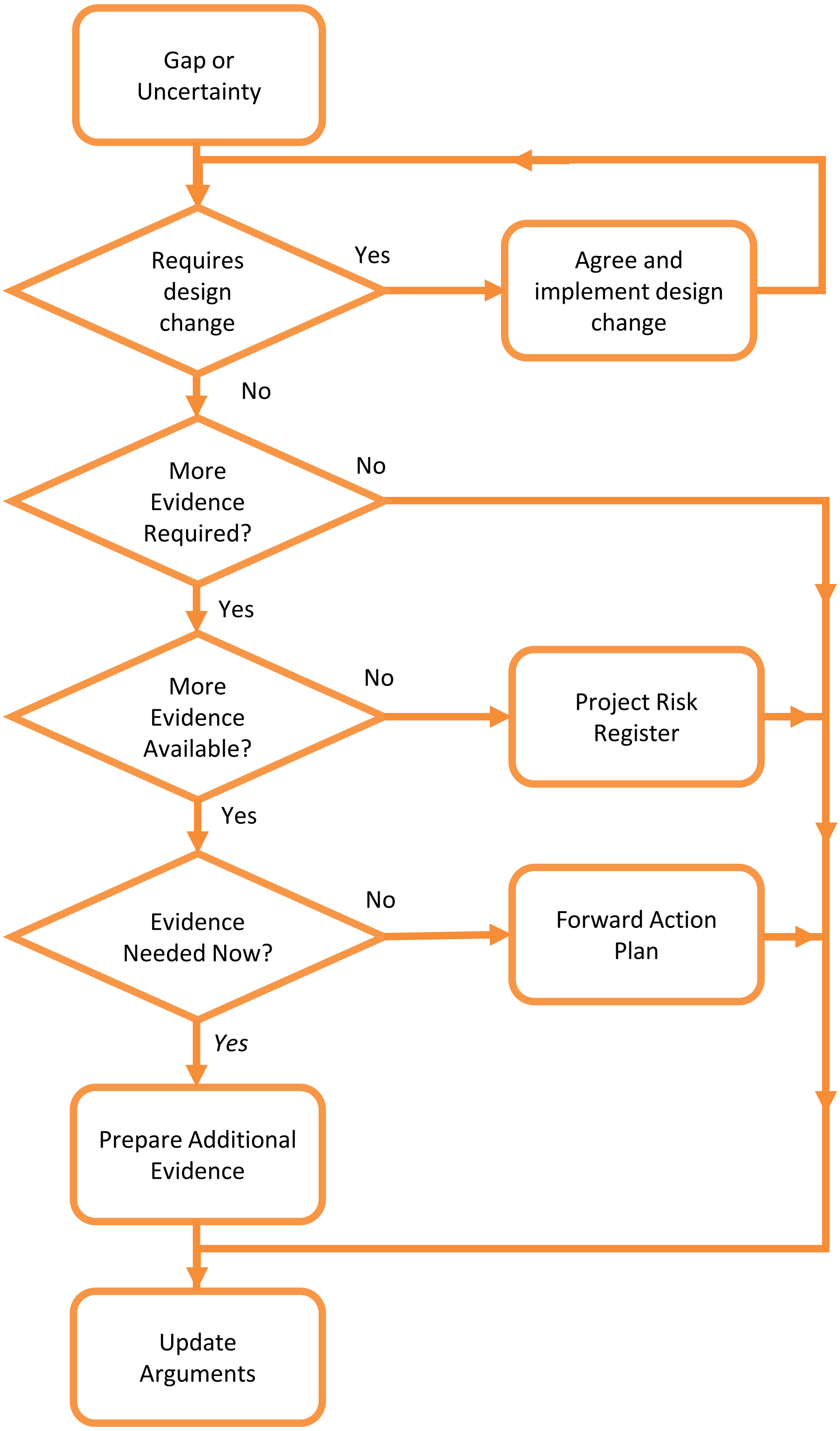


Figure 4.4-1: The Decision Tool

Following the identification of a gap or uncertainty in the design, Arguments or substantiating Evidence, the decision tool is used to determine how they should be managed. During the development of the Demonstration of BAT report gaps and uncertainties were identified and then recorded. The Hitachi-GE subject matter experts then reviewed the gaps and uncertainties and either:

- provided further design basis information;
- instigated further assessment (e.g. options assessment) which had the potential to result in a design change;
- recorded the gap as a Forward Action (FA) to be addressed at a later, more appropriate, stage of the project; or
- entered the gap or uncertainty onto the project risk register.

The additional information or assessment is then used to further develop the BAT Arguments.

A specific example of where a gap was identified and addressed is the stack monitoring requirements. Improvements relating to the location of the stack monitoring equipment were identified during a BAT workshop which resulted in an assessment. This assessment work resulted in a change to the design of the UK ABWR. Further details relating to this example are provided within the 'Approach to Sampling and Monitoring' report [Ref-20].

5. Claims, Arguments and Evidence

This section sets out the Claims, Arguments, Evidence developed for the UK ABWR through application of the methodology outlined in [Ref-3].

For the demonstration of BAT, Hitachi-GE defines a Claim as:

- a clear statement of what will be achieved; and
- a demonstration of compliance with the requirements of the P&ID and the Environmental Permit conditions that relate to BAT.

Arguments are presented to demonstrate that a Claim is valid. An Argument is further developed by:

- identifying those aspects of a design that contribute to the generation and discharge/disposal of radioactive waste;
- establishing waste streams and arisings; and
- understanding what is required to demonstrate compliance with relevant permit conditions.

Evidence is information available to support the demonstration that BAT is being applied. Evidence is required to:

- underpin arguments;
- allow examination and challenge; and
- identify key gaps (uncertainties).

Figure 5-1 presents the Claim, Argument, Evidence structure that is used in the Environmental Optimisation Report for the UK ABWR.

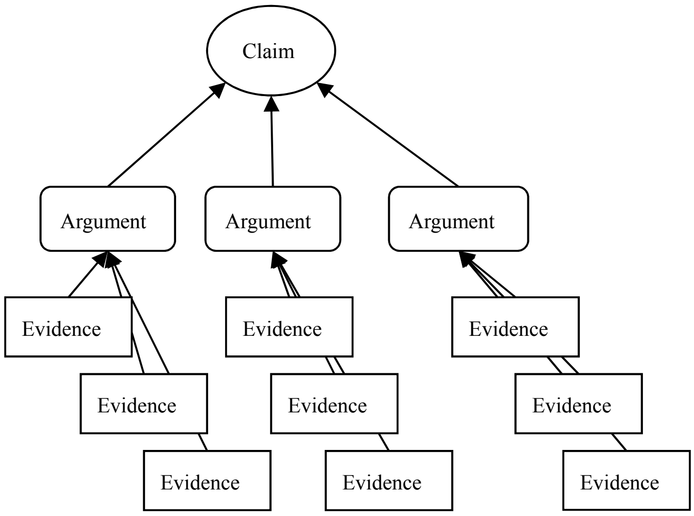


Figure 5-1: Claim-Argument-Evidence structure

Five claims have been developed that set out what Hitachi-GE need to achieve in terms of compliance with those conditions of an environmental permit that require the application of BAT. The claims have been developed under the following titles that form the main structure of the report:

- Claim 1: Eliminate or Reduce the Generation of Radioactive Waste
- Claim 2: Minimise the Radioactivity in Radioactive Waste Disposed to the Environment
- Claim 3: Minimise the Volume of Radioactive Waste Disposed of to Other Premises
- Claim 4: Selecting the Optimal Disposal Routes for Wastes Transferred to Other Premises
- Claim 5: Minimise the Impacts on the Environment and Members of the Public from Radioactive Waste that is Disposed of to the Environment

Each claim is validated by a number of arguments which taken together set out how the UK ABWR has been optimised. The evidence provided has been used to both support the development of the arguments and to substantiate the arguments. The claims, arguments and evidence set out in this document are considered by Hitachi-GE to comprise the BAT Case for the UK ABWR.

5.1. Claim 1: Eliminate or Reduce the Generation of Radioactive Waste

The generation of radioactive waste during the operation of the UK ABWR is undesirable due to the potentially harmful effects of exposure to members of the public and the environment during storage, disposal or discharge and the time, trouble and cost incurred in its management. The evolution of the UK ABWR design has sought to avoid the generation of radioactive waste at source. Where this has not been practicable, efforts have been made to minimise the activity and quantity of radioactive waste that will require subsequent management and disposal by permitted means.

The Arguments presented in support of this Claim are considered to demonstrate compliance with the standard BAT conditions [Ref-2]:

- Environmental Permit Condition 2.3.1 'The operator shall use the best available techniques to minimise the activity of radioactive waste produced on the premises that will require to be disposed of on or from the premises.'

This is also considered to fulfil the following requirement of the P&ID [Ref-1]:

- Preventing and minimising (in terms of radioactivity) the creation of radioactive waste.

The UK ABWR design contains a range of features that contribute to the substantiation of this Claim including:

- the design, manufacture and management of nuclear fuel to minimise the potential for a release of Fission Products (FP) from the fuel into the steam circuit or cooling pool water;
- the elimination or reduction of materials that are susceptible to activation at all stages of commissioning and operation;
- the reduction of the generation of Spent Fuel (SF) and Higher Activity Waste (HAW) for a given energy output;
- the reduction of the generation of Lower Activity Wastes for a given energy output;
- the prompt detection and management of failed fuel; and
- the introduction of techniques to be used during commissioning, start-up and shutdown to minimise the incidence of Stress Corrosion Cracking (SCC) of key reactor components.

In developing the Arguments presented to demonstrate the validity of Claim 1 the REPs have been taken into account. The following REPs are considered to be specifically relevant to this Claim:

- **Principle RSMDP3** ‘the best available techniques should be used to ensure that production of radioactive waste is prevented and where that is not practicable minimised with regard to activity and quantity.’
- **Principle ENDP1** ‘The underpinning environmental aim for any facility should be that the design inherently protects people and the environment, consistent with the operational purpose of the facility.’

5.1.1.Argument 1a: Design, Manufacture and Management of Fuel

The fuel assemblies present the largest source of radionuclides that are created as a result of nuclear fission in the reactor. Collectively these radionuclides are referred to as Fission Products (FP). Any release of FPs from the fuel into the steam circuit or cooling pool water have the potential to become radioactive waste that will ultimately require treatment and/or discharge to the environment. Ensuring that these FPs remain in the fuel and its cladding is a key element of the design and operation of the UK ABWR.

The manufacturer of the fuel for the UK ABWR, Global Nuclear Fuel (GNF), is engaged in a long-standing and comprehensive programme of work to improve the performance of its products and to reduce the frequency of fuel failures. Fuel failures are typically small cracks in the fuel cladding which allow FPs to be released into the steam circuit. Developments include the introduction of new filters within the fuel assembly to remove debris that can damage the fuel (5.1.1.2 Evidence: Debris Removal), quality control improvements to reduce failures at the Pellet Cladding Interaction (PCI) which can result in the cracking of the fuel cladding leading to a release of the FP from the fuel (5.1.1.3 Evidence: PCI Reduction), and the introduction of a pure zirconium liner to reduce SCC (5.1.1.6 Evidence: Selection of Fuel Cladding Materials). These improvements have been accompanied by the production of guidance to the users of GNF's fuel which clearly defines the operating parameters of the fuel and the means by which fuel failures can be minimised during operating cycles in the reactor (5.1.1.5 Evidence: Manufacturer's Guidance of Fuel Use).

GNF collaborates closely with its customers to monitor the performance of its fuel and to understand the mechanisms that give rise to fuel failures. Comprehensive data are available on the performance of its fuel in reactors in Japan, the United States of America and Europe. Analysis of these data has been undertaken by GNF which concludes that GNF's improvement programme has significantly reduced the frequency of fuel failures within Light Water Reactors (LWR) (5.1.1.1 Evidence: Analysis of Recent Fuel Failures). The data gathered from operational experience is fed back into the GNF fuel programme and is used to support the development of future enhancements.

GNF's fuel cladding is manufactured from a zirconium alloy known as Zircaloy 2. This material is widely used in the nuclear industry and has been selected because it is transparent to neutrons, resistant to corrosion and is impermeable to the migration of FPs.

Any uranium on the external surfaces of the fuel, referred to as 'tramp uranium', has the potential to undergo nuclear fission and to generate FPs that will enter the steam circuit. GNF has developed quality assurance (QA) processes that minimise the potential for the external surfaces of its fuel to become contaminated with uranium during manufacturing processes (5.1.1.7 Evidence: Manufacturing and QA Processes to Minimise Tramp Uranium).

Hitachi-GE has developed fuel handling equipment that minimises the potential for damage during transportation, loading, unloading and storage of fuel and SF. Operational experience and feedback from the operating fleet of BWRs in Japan, United States of America and Europe has shown that there is a very low frequency of fuel damage associated with the management of fuel and SF outside of the reactor (5.1.1.8 Evidence: Fuel Handing Equipment - Operational Experience and Feedback).

Collectively these measures will ensure that the transfer of FPs from the fuel to the steam circuit and the cooling pool water will be minimised and that BAT is being applied to the design, manufacture and management of nuclear fuel. This in-turn will minimise the quantity of secondary waste that is generated from the management and treatment of FPs in the gaseous and aqueous effluents.

5.1.1.1. Evidence: Analysis of Recent Fuel Failures

GNFs evolutionary product introduction strategy develops and implements new products and processes that deliver improved fuel performance. The introduction of these design changes has delivered a steady improvement in fuel reliability (Figure 5.1.1.1-1) whilst maintaining design and fabrication-related performance. In the past three decades, fuel reliability has improved by approximately three orders of magnitude. This is based on the instances of leakage of FPs from fuel rods being reduced from over five hundred rods per million operating, to below ten. In previous decades, most plants experienced at least one fuel failure during each cycle. Most fuel failure events (almost 80 percent) have been due to debris fretting, but the introduction of an improved debris filter lower tie plate (Defender) has resulted in a reduction in failures from debris by approximately a factor of 5 in GNF 10×10 fuel. Current practice is for most GNF fuelled plants to operate without a leakage for a number of years; with only a very small minority of GNF customers' plants experiencing failures in any cycle.

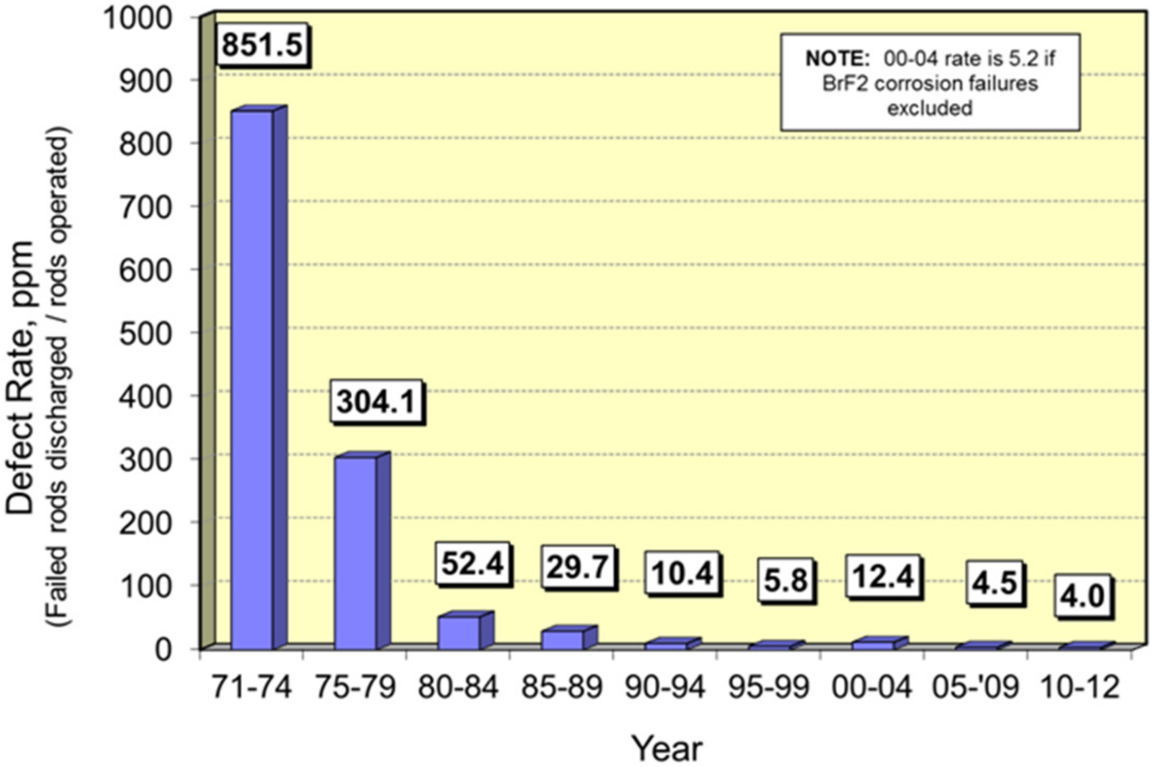


Figure 5.1.1.1-1: GNF Historical Fuel Reliability Performance

Fleet-wide, GENUSA (a GNF and Enusa Industrias Avanzandas (ENUSA) partnership) drove the improvement in reliability by systematically identifying and characterising failure mechanisms through poolside and hot cell examinations, and subsequently eliminating such failures through changes in design and fabrication of the fuel. The improvements that have been introduced as a result of this work include:

- improved pellet fabrication in the 1970's to eliminate primary hydride failures which resulted in the fuel cladding failing;
- fuel duty operating recommendations in the 1970's, followed by GE's invention and patent of zirconium-lined barrier fuel in the early 1980's, to mitigate PCI;
- corrosion-resistant cladding, with a chemistry and microstructure specifically targeted to protect against Crud-Induced Localised Corrosion (CILC) failures;
- improved cladding and welding fabrication and inspection techniques;
- tightened pellet missing surface specifications to add margin to "duty-related" failures;
- a debris filter to reduce debris related damage to the fuel rods;
- subsequent advanced debris filter designs to improve resistance to debris ingress; and
- operating guidance to maximise the capacity factor while minimising the potential for "duty-related" failures associated with power increases after control rod withdrawals.

The current performance trends lead to the following conclusions:

- In the 10x10 fuel type which will be used in the UK ABWR, the leading failure mechanism has been debris fretting which is where debris becomes entrained within the coolant and can damage the fuel assemblies. The only other significant failure mechanism observed has been several "duty-related" (PCI-type) failures, from a small number of plant manoeuvres in over 15 years of operating experience. Almost no fabrication related defects are known to have occurred in the past ~15 years of production, which represents over 7 million fuel rods. (In 2010, a single failure occurred at a plant in Spain which may be related to a fabrication defect from the ENUSA facility. A missing pellet surface (MPS) issue, or some other undetected defect, may have been a factor in a failure in Olkiluoto-1 in May 2010.).

- Today's fuel, with its increased performance capability, has the capability to be operated at increased power levels and for longer durations, both to achieve improved fuel cycle economics and to meet high-energy cycle demands. Power densities, capacity factors, cycle lengths and resultant cycle energies continue to increase and drive fuel duty. This provides operators with the opportunity to replace less fuel during an outage. Refuelling batch sizes approach 50 percent for some plants, exposing fresh fuel to conditions not typical in the past (e.g., higher duty and, in some applications, high control early in life).
- Advanced debris filters have performed well to date, with lower failure rates achieved relative to prior designs.

5.1.1.2. Evidence: Debris Removal

Debris can become entrained within the coolant and can damage the fuel assemblies which can result in a release of FPs. Debris fretting occurs when various types of debris in the coolant protrude through the lower tie plate (LTP) and cause through wall fretting of the cladding.

In 1990 GNF offered its first intra-bundle debris protection by introducing an LTP that has an entry hole that is one third the size used in the previous LTP design. This reduced the size of debris that could enter the bundle.

In 1996, GNF introduced a debris filter LTP that reduced the size of the debris that could enter the bundle by another factor of three. This filter was offered as an option on 9x9 and 10x10 products at that time. As part of GNF's zero leaker initiative, it was integrated into the GE14 fuel type as a standard feature in 2001.

The next iteration of the design was to include a debris shield. The debris shield further reduced the size of the debris that could enter the bundle relative to the first generation 10x10 debris filter, with no pressure drop penalty. The debris shield was a perforated metal plate located at the top of the LTP and was held in place by the fuel rods. The design began operation in early 2005. Field experience with the debris shield included 8 reloads and ~2,250 bundles, with only one failure.

GNF then developed a next generation "Defender™" filter (Figure 5.1.1.2-1) that further reduced the size of debris, but specifically targeted wires or wire-like debris that have been associated with cladding perforations. The first reload began operation in a United States of America (U.S.) reactor in 2006; the experience as of January 2013 includes ~75 reloads and > 15,000 bundles. Generally, a debris filter has a potential impact on channel pressure drop. The Defender has adopted a non-line-of-sight design which efficiently captures wire debris during operation with no impact on channel pressure drop.

Table 5.1.1.2-1 provides the performance for the first generation 10x10 cast debris filter LTP vs. the more recent filter designs. To date, advanced filters have a failure rate about a factor of 5 to 6 better than the original standard filters.

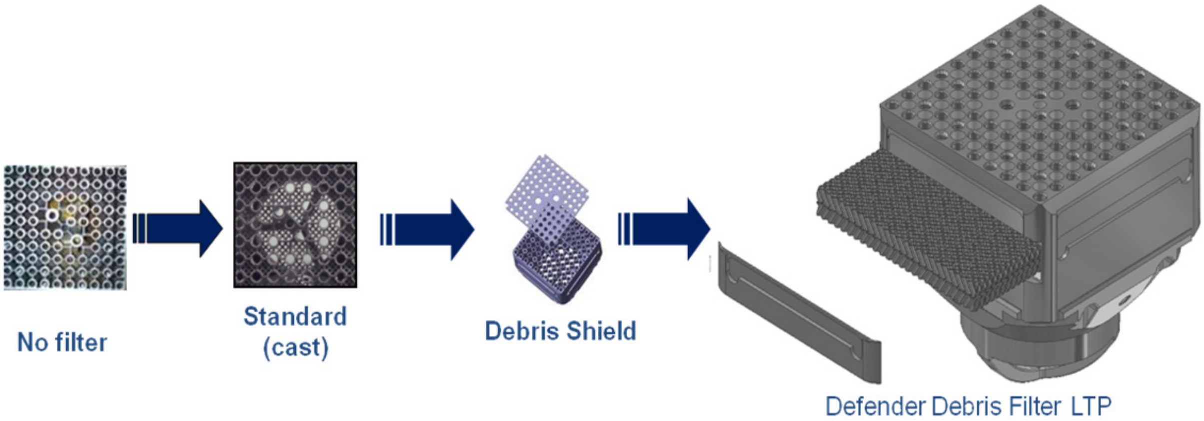


Figure 5.1.1.2-1: Development of Defender Filtration Technology

Table 5.1.1.2-1: Failure Rate of Bundles with Advanced Filters

	Standard	**Debris Shield**	**Defender**	**Advanced Filters – either Debris Shield or Defender**
Bundles	to 22,900	to 2,250	to 15,200	to 17,500
Failure Rate (failed bundles per thousand bundles)	2.9	0.5	0.5	0.5

5.1.1.3. Evidence: PCI Reduction

There are several features available to GNF that have been incorporated into the bundle design that contribute to the mitigation of PCI-type failures.

Zirconium-lined barrier fuel cladding was introduced in reload quantities in the mid-1980's as a material solution to PCI-type failures. Power ramp tests (Figure 5.1.1.3-1) and reactor fleet trials demonstrated that a zirconium lined fuel rod was highly effective in mitigating PCI-type failures. GNF fuel designs have continued to employ barrier cladding since then. Barrier cladding provides significantly improved PCI resistance, especially when combined with other mitigation strategies such as core loading and bundle promotion practices. Core loading and bundle promotion practices consist of moving fuel bundles to different locations within the reactor core at each outage in order to improve the cores efficiency and manage the burn up rates of the different fuel bundles. Because a small number of PCI-type failures have occurred in barrier fuel, operating guidelines have been implemented throughout the BWR fleet, mainly to avoid specific types of operation (for example, very long control intervals followed by step increases in power) that have been correlated with the failure events. Zirconium-lined barrier fuel cladding reduces the pellet area, but does not have a significant effect on uranium inventory. The barrier liner is thin, but is still effective enough for preventing PCI-type failure.

PCI-like failures have been correlated to fuel rods with chipped pellets (or areas of MPS) These defects can increase the probability of a PCI-type defect because there is an additional localised bending stress in the vicinity of the MPS, over and above the stress from the rod pull/power increase. Since the mid 1990's GNF has adopted improved manufacturing and QA processes that reduce the probability of chipped pellets ending up in fuel rods. The recent introduction of larger chamfered pellet edges has further reduced the likelihood that pellets will be damaged during the manufacturing process and improved pellet inspections have been implemented to identify and subsequently reject damaged pellets. Fuel pellet quality is high, and correspondingly, the fuel rod failure rate in the fleet due to PCI-type defects is at an all-time low level.

ABWR core design and operating strategies have been developed that minimise the risk of PCI related fuel failures. These strategies include:

- control cell core loading pattern;
- control rod sequence exchanges at regular operating intervals;
- operating at a flat hot excess reactivity trajectory through the cycle that minimises control rod movement; and
- the use of Reactor Internal Pumps (RIP) to facilitate power manoeuvring.

As a result of applying the above improvements, from the mid-1990s to late 2003, PCI-type failures were observed only in a few legacy 8x8 bundles (and these were found to be in those plants that had not adopted the operational guidance suggestions). No PCI-type failures were observed in the 2 million 9x9 rods delivered from the GNF factories since the introduction of these products in 1990. In the period from late 2003 to early 2007, and in 2010 at one European plant (Olkiluoto-1), GNF experienced several PCI-type failure events in 10x10 fuel. All PCI-type failures in 10x10 barrier fuel occurred in a small number of reactor manoeuvres. The total failure rate in 10x10 barrier fuel due to PCI mechanisms is less than 4 ppm.

5.1.1.4. Evidence: Manufacturing Improvements

Over the past ten-year period, approximately 4,000,000 GNF 9x9 and 10x10 fuel rods have been fabricated and placed in operation. Within this population, GNF has identified a single fuel rod failure caused by a fabrication defect. GNF states that this performance is the result of the safeguards and process improvements that have been implemented since the identification of several factory-related failures in fuel fabricated during the 1980s. A tighter MPS specification was established in 1995 that eliminated the flaws associated with some PCI failures. Multiple inspections were established to assure compliance with the MPS specification to ensure that fuel pellets did not have missing surfaces which could then lead to fuel failures during operation. A state-of-the-art tubing inspection system, called Multi-Inspection and Data Acquisition System (MIDAS) was introduced that provided a thorough clad identification and Outer Diameter (OD) inspection process. It specifically addressed the OD flaws thought to be associated with some past duty-related failures. This system provides two independent flaw inspections: 100 percent of the OD by eddy current (OD only) for gross flaws as well as some geometries difficult to assess via ultrasonic; and >100 percent coverage of the OD and identification by ultrasonic, which is typically capable of identifying flaws in the order of 25 to 50 microns (one to two mils).

MIDAS also uses Ultrasonic Testing (UT) to measure critical dimensions along the full length of the fuel rod, since these directly affect such fuel performance behaviours as stress and bow. Fuel rods that do not meet specifications for dimensions or flaw criteria are automatically rejected. The MIDAS record for each rod is recorded in the Fuel Business System (FBS) and references to a barcode identifier laser-marked on each rod. FBS will not allow a reject tube to be processed into fuel rods and ultimately fuel bundles.

Again, learning from past experience, the GNF fuel rod process requires a 100 percent UT inspection of both the upper and lower end plug weld of the fuel assembly. All historical weld defects are targeted, including tungsten or other foreign material inclusions, inadequate bonding or penetration, and grain boundary separation. Weld records are tied to the fuel rod via its barcode and entered into GNF's FBS. The system will lock out any weld rejects, precluding further processing.

Whilst GNF has been successful in delivering improvements in fuel design resulting in a reduction in the number of potential manufacturing-related failures, GNF recognises that it must continue to drive for manufacturing excellence to assure similar performance in the future. Several areas GNF is focused on are described below:

- Debris remains the number one cause of failures, and it is imperative that bundles delivered from GNF's factory be free from debris. Many actions have been taken to assure success in this area, including improved inspections and the establishment of debris exclusion zones in the bundle assembly and packing areas. An additional improvement GNF has implemented is the stainless steel fuel-shipping containers, replacing the wooden container, which often presented a risk of paint chips.

- GNF has deployed new grinding stations that also include, via a feedback system, the 100 percent OD inspection of each pellet. This system will 1) verify that each pellet meets dimensional specifications, 2) feed back to the grinder any necessary adjustments to maintain process control, and 3) inspect the pellets for OD surface defects. This represents a very significant upgrade to the overall pellet fabrication process.

Further detail on the fuel assembly design is provided in the Pre-Construction Safety Report (PCSR) Chapter 11: Reactor Core [Ref-88].

5.1.1.5. Evidence: Manufacturer's Guidance of Fuel Use

GNF fuel is operated in accordance with a variety of design and licensing limits, such as Thermal-Mechanical Operating Limits (TMOL) for linear heat generation rate (kW/ft. or Watts/cm) as a function of exposure, and limits on exposure in terms of bundle average, rod average, or peak pellet, which can vary by the country in which the fuel is licensed. There are specifications for dry or wet storage prior to irradiation, including water quality. For PCI-type failure mitigation, power manoeuvring guidelines are provided which propose exposure-dependent threshold power levels above which power increases should occur at or below certain rates. These guidelines help to mitigate the tensile stress of the cladding (i.e., limiting the rate of pellet thermal expansion) and the release of embrittling FPs (iodine in particular) that promote SCC (i.e., limiting the rate of pellet temperature increase), both of which are key factors contributing to PCI-type failures. Implementation of these manoeuvring guidelines may limit the operating condition, but it has been shown to have a negligible impact on plant capacity factors in today’s BWR fleet.

5.1.1.6. Evidence: Selection of Fuel Cladding Materials

GNF offers a unique version of the zirconium alloy, Zircaloy-2, as its cladding material. Zircaloy-2 is widely used as a cladding material in BWRs, but GNF’s cladding has an inner liner of pure zirconium with additions of iron for corrosion resistance in order to serve as a buffer between the Zircaloy-2 and the swelling of the uranium pellet. The softer liner has been effective as a barrier for PCI since the early 1980’s. The outer cladding is annealed in order to achieve the final state of full recrystallisation.

In BWRs, a major performance deterrent for the cladding is nodular corrosion that occurs due to exposure to the reactor environment. GNF has developed a substantial database reflecting the performance of its cladding which is exposed to today’s modern water chemistries, including hydrogen and zinc injections as well as noble metal applications. The cladding may also be exposed to reactor water chemistry variations experienced within some plants. Based on this growing experience base, GNF has reached the following conclusions:

- Should the cladding be breached due to debris fretting at spacer grid locations, the corrosion resistant liner (mentioned above) is an effective solution to post-failure degradation. It appropriately balances the need for the cladding liner to provide both PCI protection as well as corrosion resistance.
- Modern cladding designs are required to have excellent corrosion resistance to high exposures even in today’s reactor environments, with modern water chemistries.

These conclusions have led GNF to develop and introduce the current Zircaloy-2 barrier cladding, known as Process 9 cladding, which combines the best features of previous designs, while taking steps at both the tube shell supplier and GNF’s Wilmington tube fabrication facility to reduce process variability.

Process 9 utilises modifications to the alloying elements in order to tighten variability in the ingot chemistry and enhance corrosion resistance. Tightening and biasing the alloying elements of iron, tin, nickel and chromium when compared to standard American Society for Testing and Materials (ASTM)

Zircaloy-2 specifications has proven to demonstrate better performance in nodular corrosion resistance. In the raw material manufacturing process, a rapid quench from the beta metallurgical phase at the billet stage, known as hollow billet beta quench, provides a smaller second phase particle size distribution and a more uniformly reproducible product. Tubing made from hollow beta-quenched billets has excellent corrosion resistance in laboratory corrosion tests and in the field. This was one of the earliest corrosion enhancing processes implemented.

5.1.1.7. Evidence: Manufacturing and QA Processes to Minimise Tramp Uranium

Tramp uranium is uranium or uranium dioxide dust that clings to the outside of the fuel elements and is insufficiently cleaned off during fabrication. Once in the reactor, it will undergo fission and its FPs readily enter the reactor coolant.

A number of measures have been introduced to minimise uranium oxide contamination on fuel rod surfaces. The handling process for unsealed uranium oxide (e.g., pellet loading process and upper end plug weld process) has the potential to spread uranium oxide contamination onto the fuel rod surface. The unsealed uranium oxide is exhausted by air conditioning equipment to remove any uranium dust that becomes entrained in the air. In over 99 percent of cases, the measurement result for uranium oxide contamination on the fuel rod surface is lower than the detection limit. QA measures ensure that tramp uranium remains below the QA thresholds.

In order to minimise uranium contained in the Zircaloy raw material, GNF's current material specification is for less than 3.5 ppm uranium in zirconium alloys. In practice the material certification reports indicate that it is usually reported as <0.5 ppm. The uranium content is driven by the input material for Zirconium processing, which is sand. A part of the processing from sand to crystals leaves the uranium in the silica waste stream; separation processing takes the majority of the residual uranium to the hafnium side. GNF's Zircaloy tube shell suppliers report that the hafnium content in the feed has been significantly reduced since implementing the processing change with the intermediate step going to crystals.

Additional controls during Zircaloy processing include preventing the use any Zircaloy that has been irradiated. Use of returned Zircaloy-2 tubing from rods scrapped during manufacturing that had been loaded with pellets can also result in traces of pellet material if the recycled ingot includes material melted from the tubing. Material from these scrapped rods is prevented from entering the manufacturing process.

The low levels of impurities (low ppm levels) result in a very low level of off-gas FPs activity, even for initial cores with no fuel failures. GNF's monitoring of off-gas activity data in the fleet supports the observation that uranium impurities in the cladding are at the lowest levels they have been in BWR history. Many cores today that have not had a fuel failure in a decade or more, have extremely low "tramp uranium based" off-gas activity levels, less than 3.7 MBq/s is routinely achieved, and some plants observe less than 1.85 MBq/s.

5.1.1.8. Evidence: Fuel Handing Equipment - Operational Experience and Feedback

Since 1974, Hitachi-GE has manufactured and supplied the Fuel Handling Machine (FHM) for BWR, ABWR and fuel reprocessing plants. The total number of the manufactured and installed FHM's is 20. No fuel damage or collision of fuel during fuel handling operations has occurred. Operational experience validates the effectiveness of the FHM design and systems that preclude the dropping of a fuel assembly.

5.1.2. Argument 1b: Reactivity Control

Fluids and materials that pass through the reactor core are exposed to an intense field of neutrons generated from nuclear fission. Interactions with neutrons in some instances result in the generation of activation products which require treatment and/or disposal as radioactive waste. The elimination or reduction of materials that are susceptible to activation is important to the minimisation of radioactive waste produced in the UK ABWR.

The UK ABWR uses a physical method, known as 'recirculation flow control' for controlling the rate of the nuclear reaction in the core (reactivity) (5.1.2.1 Evidence: Recirculation Flow Control). The ratio of water to steam is managed by controlling the recirculation flow rate in the core. The water acts as a neutron moderator and increases the rate of fission. Put simply, the greater the ratio of water to steam in the reactor core, the higher the reactivity will be and vice versa. The UK ABWR also uses burnable gadolinia (Gd_2O_3), a neutron poison, for reactivity control (5.1.2.2 Evidence: Gadolinia). The most effective burnable poisons must be depleted in one operating cycle so that no residual poison exists that could impact on the future fuel cycle length. It is also desirable that the positive radioactivity from poison burn-up matches the almost linear decrease in fuel reactivity from FPs build-up and uranium-235 depletion. Gadolinia which is a commonly used burnable poison in BWRs has been selected as the concentration can be controlled so that the poison is essentially depleted during the operating cycle.

Control rods are used to absorb neutrons and by changing their location are used to control the reactivity in part or all of the reactor core. There are two types of control rod; boron carbide (B_4C) rods and hafnium rods (5.1.2.3 Evidence: Introduction of Hafnium Control Rods). The UK ABWR is likely to use both boron carbide control rods and hafnium control rods in its reactor. Hafnium control rods have a longer operational life and are used not only for shutdown but for reactivity control during normal operation, which results in higher exposure of control rods. Boron carbide control rods are normally used only for shutdown. Hafnium control rods reduce the quantity of waste generated from the less frequent replacement and management of control rods at the end of their service life.

The use of the recirculation flow control method, burnable gadolinia and control rods eliminate any need for chemical agents to control the reactivity. In Pressurised Water Reactors (PWR) chemical agents such as boron are used to control reactivity and are a significant source of the tritium that is discharged to the environment. Reactivity control using 'recirculation flow control' compared with boron based water chemistry, eliminates a significant source of tritium. Optimising the generation of tritium is considered BAT as there are no large scale economic systems for the treatment of tritium and the practice across the nuclear industry world-wide is that tritium is managed by discharge to the environment. Discharge data in Japan shows that the amount of tritium disposed of to the environment from (A)BWRs is lower than PWRs (5.1.2.4 Evidence: Reactivity Control – Operational Experience and Feedback).

5.1.2.1. Evidence: Recirculation Flow Control

Reactivity within the UK ABWR is predominately controlled by changing the volume of water, which is a moderator, within the reactor. Simplistically, increasing the flow of water using the RIPs will increase the volume of moderator in the reactor consequently increasing reactivity. This increases steam production and therefore the power output. Inversely reducing the flow of water using the RIPs will decrease the volume of moderator in the reactor and therefore reduce reactivity, steam generation and the power output. In comparison PWRs are single phase which means that they are fully moderated. Reactivity is controlled by the addition of boron, which is a neutron absorber. The nuclear reactions involving boron and associated chemicals result in the generation of tritium in PWRs [Ref-7].

Recirculation flow rate is variable over a range from a minimum pump speed flow of 20% to the flow required to achieve rated core power. The flow control range allows automatic regulation of reactor power output without the requirement to move the control rods [Ref-8].

5.1.2.2. Evidence: Gadolinia

Following decades of manufacturing and operational experience gadolinium (III) oxide (Gd_2O_3), commonly referred to as gadolinia, has become the industry standard material for use as a burnable neutron absorber in BWRs. These burnable poisons are required for additional control during shutdown and for power output shaping particularly during early parts of the cycle. The most effective burnable poisons are those that become depleted in one operational cycle so that no residual poison exists that could impact on the future cycle length. It is also desirable that the positive radioactivity from poison burn-up matches the almost linear decrease in fuel reactivity from FP's build-up and uranium-235 depletion.

The fuel rods are loaded with fuel pellets consisting of either uranium dioxide (UO_2) or uranium dioxide with gadolinia (UO_2-gadolinia). The location of the UO_2-gadolinia pellets and the concentration of the gadolinia in these pellets, along with the enrichment of the UO_2 in each pellet (both UO_2 and UO_2-gadolinia) is carefully chosen to provide the desired characteristics. Additionally, the gadolinia concentration in each UO_2-gadolinia pellet is determined so that the poison is essentially depleted during the operating cycle. Gadolinia has been used in GE BWRs since the early 1970s, and has proven to be an effective and efficient burnable poison [Ref-8].

The concentration of the gadolinia contained in the uranium oxide and the number of rods that will contain gadolinia will be determined as part of optimising the reactor performance. This process will take account of relevant design policies and the reactor core performance requirements. The desired amount of gadolinia designed into a batch of fuel is dependent upon the fissile enrichment level, batch size and the operating cycle length [Ref-9].

The use of burnable poisons has the following benefits:

- reduces the number of control rods that are required for controlling reactivity during operations, which consequently reduces the number of control rods requiring disposal;
- the reduced requirement for control rod usage simplifies core operation and management for the operator; and
- burnable poisons can be distributed more uniformly than control rods and are less disruptive to the core's power distribution.

Gadolinia was selected for use in the UK ABWR over alternative burnable poisons; notably boron, hafnium and europium for the following reasons:

- Boron generates helium and tritium during the neutron absorption process. The build-up of these nuclides can cause an internal pressure rise and tritium can result in the hydride embrittlement of the fuel cladding.
- Hafnium and europium generate nuclides that also have a large absorption cross section. The negative reactivity of the burnable poison and its products does not decrease over time and therefore, such absorbers are unsuitable from the viewpoint of fuel economy.
- The residual gadolinia after burn-up is small and less than the other burnable poisons.

A disbenefit of gadolinia is its higher relative cost compared to boron. However, this is outweighed by the benefits presented above. The use of gadolinia also generates other radionuclides. However, the majority of these radionuclides have relatively short half-lives of less than one year (except Gd-152) and there is no significant radiological impact on waste categorisation, treatment and disposal when the fuel (including the burnable poison) is disposed of. Gd-152 is considered to be a stable nuclide and has no impact on dose and disposal [Ref-73].

5.1.2.3. Evidence: Introduction of Hafnium Control Rods

The control rods and control-rod drive systems provide the functionality of controlling the power from the reactor and adjusting the power distribution across the reactor core. Power control is implemented by varying the control-rod position, and the power distribution is adjusted by changing the location of the control rods.

The control rod contains neutron absorbers; either boron carbide powder filled stainless steel tube or hafnium metal. Figure 5.1.2.3-1 details the assembly of a boron carbide control rod. Hafnium control rods are similar in design but with hafnium replacing the boron carbide powder filled stainless steel tubes [Ref-74].

Table 5.1.2.3-1 shows the control rod specification for boron carbide type and hafnium type control rods.

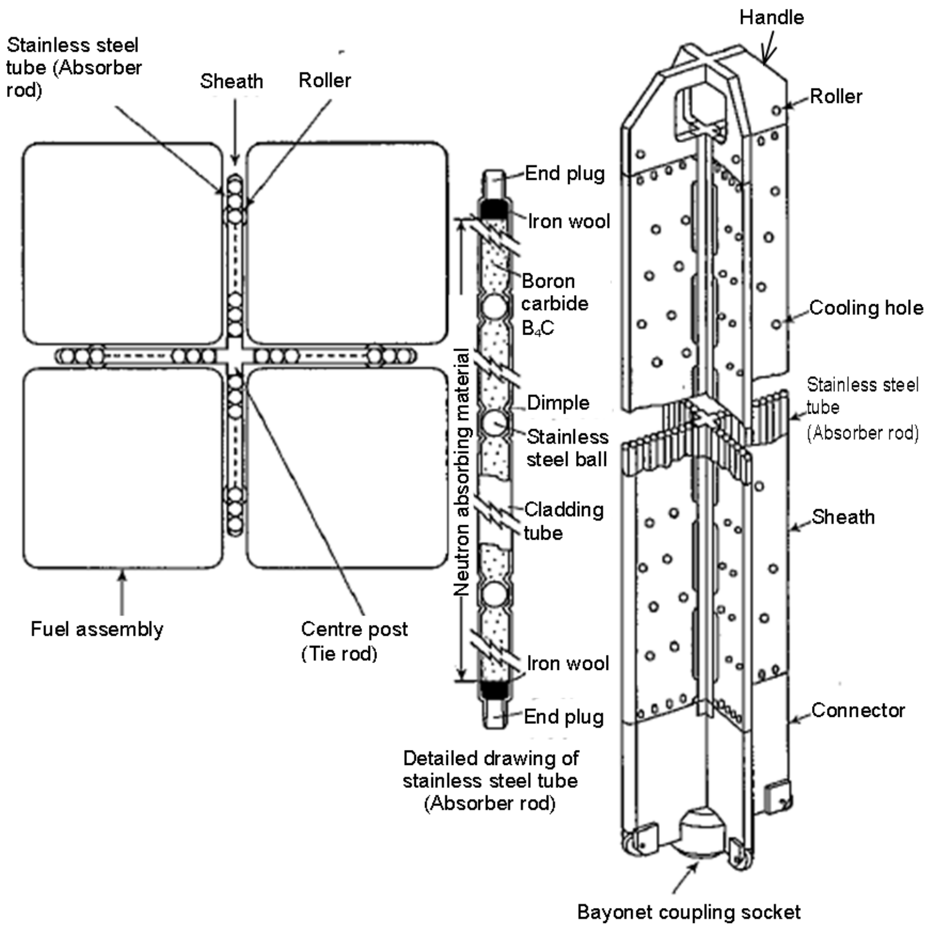


Figure 5.1.2.3-1: Boron Carbide Control Rod Assembly [Ref-74]

Table 5.1.2.3-1: Main Specification of Control Rods

Control rod	Weight (kg)	Active Absorber Length (m)	Neutron absorber	
			Absorber	Number
Type I	80	3.63	Boron carbide powder.	72 cladding tubes filled with boron carbide powder (per control rod).
Type II	100	3.63	Hafnium metal.	16 Hafnium tubes (per control rod).

Hafnium control rods will be used in the 29 control cells located in the reactor core. Boron carbide control rods will be used in the 176 shutdown cells. The purpose of the control cells is to control reactivity during power operations whilst the other control rod cells are used for shutdown operations [Ref-55]. An advantage of hafnium control rods is that they have a longer operational life than boron carbide control rods. The process of neutron adsorption in boron carbide control rods depletes the boron as illustrated in Figure 5.1.2.3-2. The boron reacts, emitting helium resulting in a reduction in the amount of boron-10.

$$^{10}\mathrm{B} + \mathrm{n} \rightarrow \, ^{7}\mathrm{Li} + \, ^{4}\mathrm{He}$$

Figure 5.1.2.3-2: Neutron Capture Reactions of Boron

Hafnium control rods are able to capture many neutrons because most of the isotopes produced by the neutron capture reactions still have the capability to capture neutrons, as shown in Figure 5.1.2.3-3. Thus, the nuclear life time of hafnium is longer than that of boron and as a result the hafnium control rods do not need replacing as frequently, resulting in the generation of less waste. Boron carbide control rods are still required for shutdown due to their large cross section of thermal neutron capture.

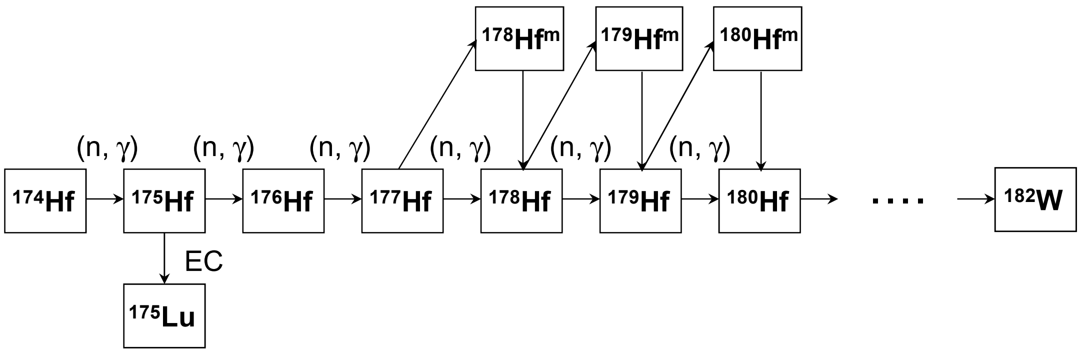


Figure 5.1.2.3-3: Neutron Capture Reactions of Hafnium

During power operation, about half of the 29 hafnium control rods located in control cells are used. The inserted control rods are exchanged at an interval determined by the operator typically following 3~4GWd/t exposure. The lifetime of a hafnium control rod is estimated to be 2 cycles with continuous irradiation, where 1 cycle is expected to be "operation + outage = 17 + 1 months". Furthermore the control rod pattern (the location where the control rods are inserted) is changed regularly during the power operation. As a result, the number of hafnium control rods replaced is estimated to be 29 per 4 cycles (72 months), which is nominally equal to 5 rods per year [Ref-55].

If boron carbide control rods were used as the control cell control rods instead of the hafnium rods during power operation, as they were in previous BWR designs, the replacement rate would be approximately double as the lifetime of a boron carbide control rod with continuous irradiation is estimated to be 1 cycle as displayed in Table 5.1.2.3-2. The use of hafnium rather than boron in the control cell control rods results in approximately 300 kg less waste per year that will require pre-treatment and disposal. Disposal of hafnium control rods has been assessed in the disposability assessment [Ref-84] and are considered to pose no disposal issues.

Table 5.1.2.3-2: Control Rod Waste per Year

Items	Hafnium control rod	Boron carbide control rod*
Control rod lifetime with continuous use	2 cycles	1 cycle
Exchange frequency	29 per 4 cycles	29 per 2 cycles

Control rod waste	5 per year	10 per year

*In the case where boron carbide control rods are used as the control cell control rods.

Further to this, the generation of helium and tritium as a result of boron-10 neutron capture results in the stainless steel tubes containing boron carbide powder becoming gradually pressurised and swollen by an increase in the internal pressure and volumetric swelling. This swelling has the potential to result in very small scale, localised cracking of the boron carbide tubes and the release of small quantities of tritium into the reactor water. Monitoring of tritium in the reactor water will detect any increase in the concentration of tritium and will enable a future operator to identify if a crack has occurred. Appropriate management of the operational lifetime of the boron carbide control rods will minimise the likelihood that cracking shall occur. However, in the event that a tube does crack discharges of tritium into the environment are not expected to increase by more than a few percent [Ref-49]. Following cracking there may also be a release of carbon-14. However this is considered to be small, 6 orders of magnitude less than the normal yearly carbon-14 discharge [Ref-82]. In hafnium type control rods, hafnium captures neutrons, and γ-rays are emitted. Thus, hafnium type control rods do not emit any gaseous products and therefore do not suffer from swelling related cracking.

Further detail on the CR is provided in the PCSR Chapter 11: Reactor Core [Ref-88].

5.1.2.4. Evidence: Reactivity Control - Operational Experience and Feedback

By using the recirculation flow control method, burnable gadolinia and control rods the UK ABWR does not require the injection of chemicals to control reactivity. Chemical controls in PWRs, through the introduction of boron and associated chemicals, produce quantities of tritium which is then disposed of to the environment. By eliminating the requirement for boron and associated chemicals the ABWR produces significantly less tritium. This is demonstrated in Table 5.1.2.4-1 which summarised discharge data from Japan [Ref-10] and the expected annual discharge from the UK European Pressurised Water Reactor (EPR) (UK EPR PCSR chapter 11.3).

Table 5.1.2.4-1: Difference in Tritium Discharge between ABWR and PWR

Radionuclide	ABWR annual discharge (Bq/y)	UK EPR expected annual discharge (Bq/y)
H-3(gas)	1.2E+12	5.2 E+13
H-3(liquid)	2.0E+11	

5.1.3. Argument 1c: Efficiency of Fuel Use

The efficiency with which the nuclear fuel is used in the UK ABWR and the frequency with which it is changed will influence the amount of SF and HAW that is generated during operations. Reducing the generation of SF and HAW for a given energy output is an important part of optimising the nuclear fuel cycle from an environmental perspective. This applies to the selection of the nuclear fuel and the final choice for its management prior to disposal. The design of the UK ABWR prevents the unnecessary generation of waste and discharges associated with its management by ensuring that the minimum quantity of SF is produced per unit of electricity generated.

The UK ABWR reactor core is arranged as an upright cylinder and contains a large number of fuel assemblies (5.1.3.1 Evidence: Configuration and Geometry of the Reactor Core). The design of the core has evolved over many years of BWR operation. Compared with a PWR, the BWR has a smaller power density, larger fuel inventory, more fuel bundles, and smaller ratio of fuel assembly exchange. The flexibility of the fuel loading pattern allows fresh fuel to be placed at the core interior and old fuel to be placed at the core periphery. In addition, each fuel rod has axial enrichment distribution with lower enrichment at both top and bottom end. These BWR characteristics lead to increased fuel efficiency due to a decrease in neutron leakage. Also, the BWR has "spectral shift operation" for saving uranium. In the BWR, the fuel with high void fraction has a high energy neutron spectrum, therefore more plutonium is built up in the fuel. This effect increases in proportion relative to the period of the burn-up with high void fraction. By taking advantage of these BWR characteristics, the core is operated at the high void fraction, from the start to the middle of the operation cycle, so that plutonium build-up is promoted. At the end of the operation cycle, the core is operated at the low void fraction so that plutonium burn-up is again promoted. The reactivity of the plutonium that is generated improves fuel efficiency.

Typically the first shutdown for refuelling will take place up to 13 months after the start of initial power operations. Thereafter the cycle length can be varied up to 18 months using GE14 fuel (GE14 fuel comprises 10x10 array of fuel rods, two water rods, spacers, an LTP, an upper tie plate and a fuel channel) with further potential to achieve 24 month cycles. The desired fuel cycle length is achieved by optimising the refuelling batch size and the average enrichment of the fuel bundles. Operational experience has demonstrated the length of each fuel cycle using GE14 fuel (5.1.3.2 Evidence: Efficiency of Fuel Use - Operational Experience).

5.1.3.1. Evidence: Configuration and Geometry of the Reactor Core

The BWR is somewhat unique in its ability to accept fuel design advancements owing to the core lattice configuration that separates the fuel assembly from the control rods. Since the initial deployment of the BWR, fuel designs have undergone numerous evolutions and reactors that were initially fuelled with 7x7 fuel bundles (i.e., bundles comprised of a 7x7 array of fuel pins) are now loaded with 10x10 bundles with substantial performance and reliability improvements. Relative to early designs, contemporary fuel bundles can generate more than twice the amount of energy with reduced enrichment requirements and demonstrate greatly increased reliability (Figure 5.1.1.1-1). Fuel technology is continuously advancing and the ABWR is designed to utilise fuel technology for the entire BWR fleet - and the fleet fuel experience is applicable to the ABWR.

In addition, the ABWR has adopted an N-lattice geometry that provides a small increase in the intra-assembly bypass gap width relative to BWR/5-6 designs. The N-lattice benefits include an increased cold shutdown margin, moderate void reactivity coefficients, and increased margin to channel/control rod interference. This allows for greater core design and operational flexibility as well as simplicity. This is reflected in the equilibrium core design for the UK ABWR shown in Figure 5.1.3.1-1. In this core the fresh fuel assemblies are distributed throughout the centre region, alongside assemblies that have already been partially burned. The lower reactivity, higher burn-up assemblies are placed in the outer peripheral region to minimise neutron leakage. The core is operated at rated power and 90 percent core flow throughout most of the cycle utilising only deeply inserted control rods, thereby promoting spectral shift operation and achieving a high degree of fuel efficiency. This optimised core design satisfies all thermal margin design limits owing to the relatively low power density of the UK ABWR, in combination with a GE14 10x10 bundle design. The reload batch size and enrichment have been selected to achieve a batch average discharge exposure of 50 GWd/MT.

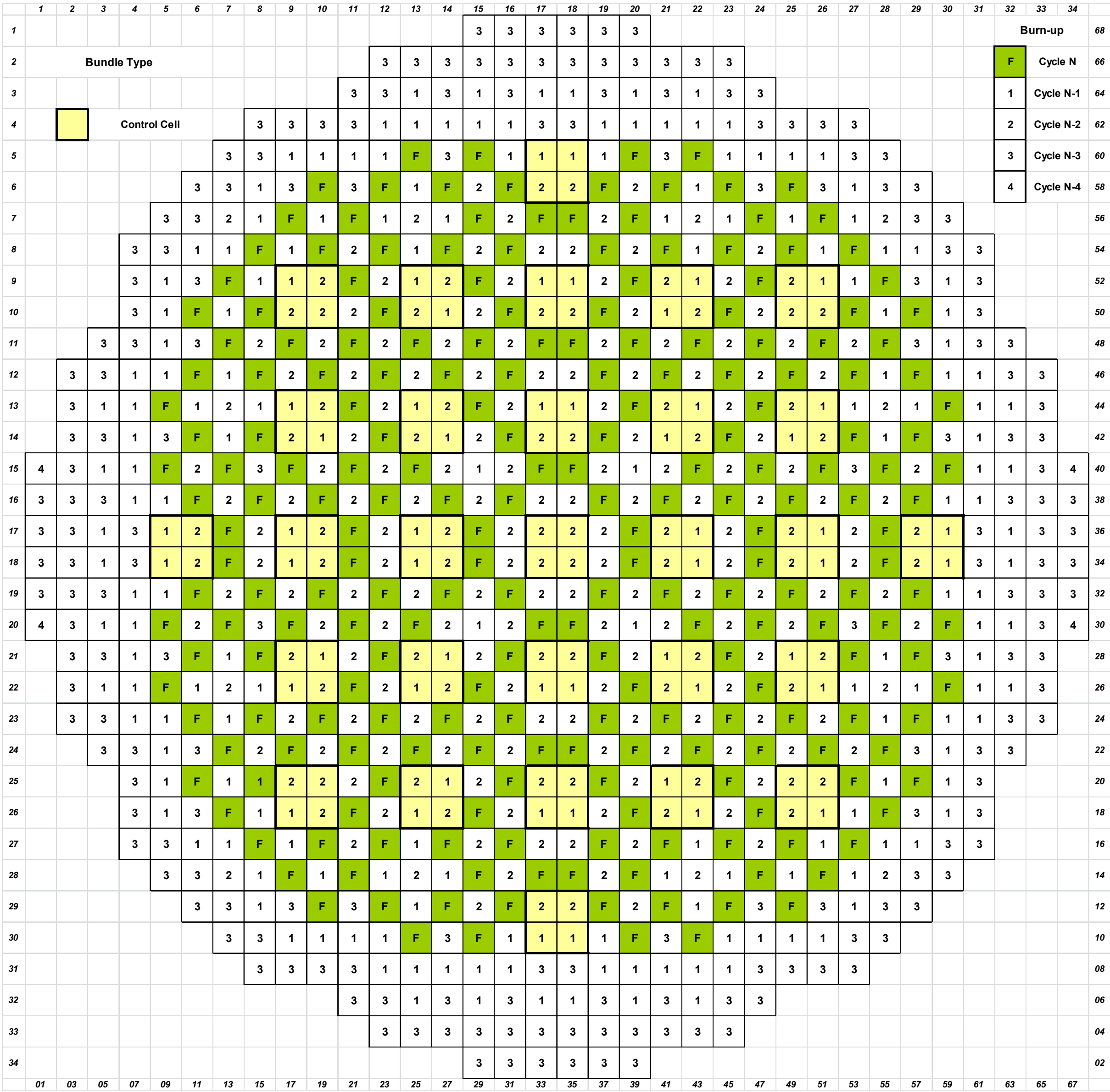


Figure 5.1.3.1-1: Reference Equilibrium Core Loading Map

5.1.3.2. Evidence: Efficiency of Fuel Use - Operational Experience

Fuel efficiency can be evaluated in several ways including:

1) the reload batch size;
2) the batch average discharge burn-up (or exposure);
3) the enrichment required for a specified discharge exposure; and
4) the amount of thermal energy produced per unit mass of fissile U235 loaded.

The ABWR with its moderate thermal power density, expanded core flow capability, large core size and favourable operating limits has the merit of maximising discharge burn-up with modest enrichment requirements, as demonstrated in the UK ABWR equilibrium core design. These features allow the ABWR to have fuel replacement requirements that are among the lowest in the BWR fleet.

A key feature of the ABWR is the large core flow range available at rated power conditions, from 90 to 111 percent, provided by ten RIPs. This core flow capability can be used to enhance a “spectral shift” operating strategy – spectral shift is an inherent BWR characteristic. In the BWR, the fuel operating in a high void fraction region (i.e., the upper core region) experiences a high energy neutron spectrum, resulting in the production of plutonium. The amount of plutonium increases in proportion to the burn-up period with high void fraction. A high void fraction can be achieved by operating at 100 percent of rated core power and 90 percent of rated core flow for an extended time. For the UK ABWR equilibrium core, operation at low flow is possible for the majority of the operating cycle. After approximately three-quarters of the operating cycle, core average axial power shifts toward the upper core region and the plutonium is utilised. Beginning and end of cycle void fraction and axial power profiles illustrating this spectral shift effect are shown in Figure 5.1.3.2-1 and 5.1.3.2-2 respectively. Contemporary fuel designs help to promote this spectral shift operating strategy, which significantly improves fuel utilisation. Fuel designs incorporate axial zoning of enrichment and gadolinia to further promote spectral shift operation by allowing a more bottom peaked axial power distribution while still satisfying thermal margins. Axial zoning also eliminates the need to insert shallow control rods for power shaping. In summary, the UK ABWR can achieve a high level of fuel efficiency by adopting an enhanced spectral shift operating strategy made possible by a 90-111 percent flow range and contemporary fuel designs.

BWR fleet history and fuel evolution also have demonstrated dramatic improvements in fuel utilisation. BWRs that were loaded with 7x7 fuel bundles during the 1970’s typically achieved discharge exposures of approximately 20 GWd/t. BWRs loaded with 10x10 fuel bundles today can achieve discharge exposures of 50 GWd/t.

Finally, using the GE14 10x10 fuel bundles, fuel cycle lengths can be varied from 12 to 24 months. Shorter fuel cycles allow for better fuel utilisation and burnup but result in more outages over the plants lifetime. Due to the inherent generation of radwaste during an outage this also leads to greater radioactive waste generation.

Longer fuel cycles result in less outages so less radioactive waste is produced but fuel utilisation and burn up is affected. This is due to there being less certainty of the core’s starting conditions (because it is based on the end conditions of the previous cycle) so slightly more conservative estimates are applied when deciding how much fuel to replace. This ultimately leads to fuel being replaced prior to reaching its maximum burn-up.

An 18 month fuel cycle is assumed on the basis that it is a compromise between the two extremes. However, as stated, the actual cycle length to be implemented will be decided by the future operator and will be decided on by balancing fuel burn up with radioactive waste produced.

In summary, the ABWR coupled with modern fuel designs can support a wide range of operating plans while achieving a high level of fuel efficiency. Advances in fuel technology have consistently demonstrated reduced fuel requirements for the BWR fleet such that the GE14 (fuel type) equilibrium fuel cycle projection for UK ABWR can be considered conservative when evaluating SF generation over the plant lifetime.

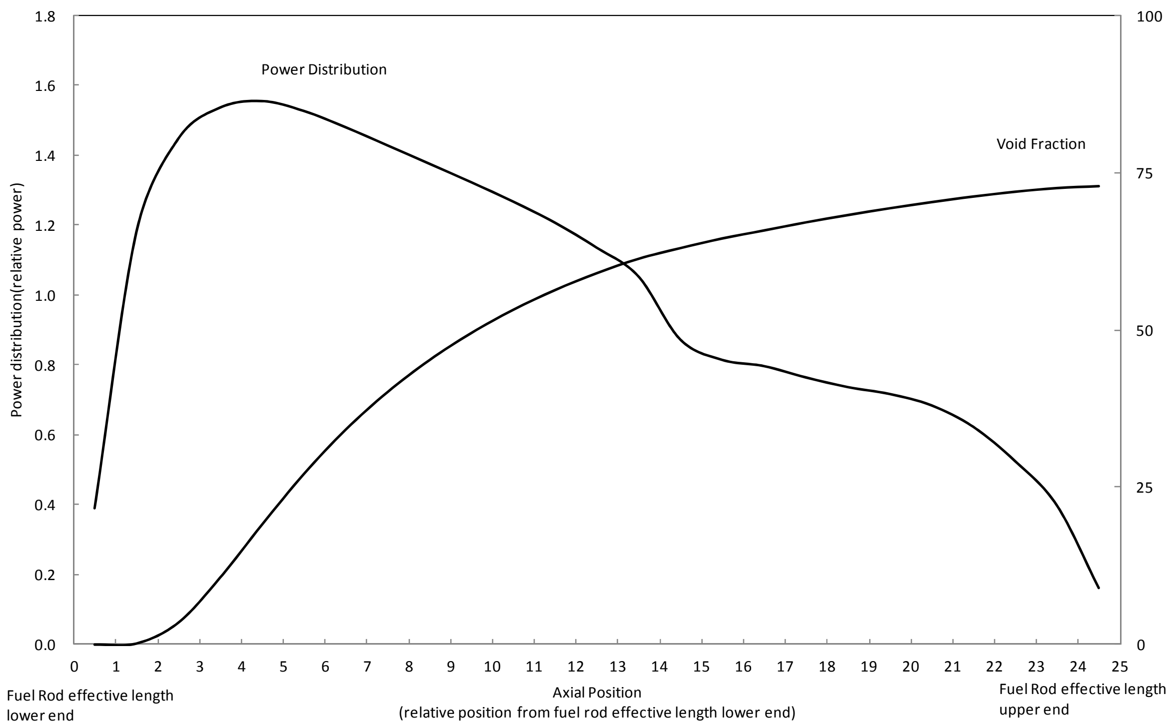


Figure 5.1.3.2-1: Axial Power and Void Fraction Distribution (Beginning of Cycle)

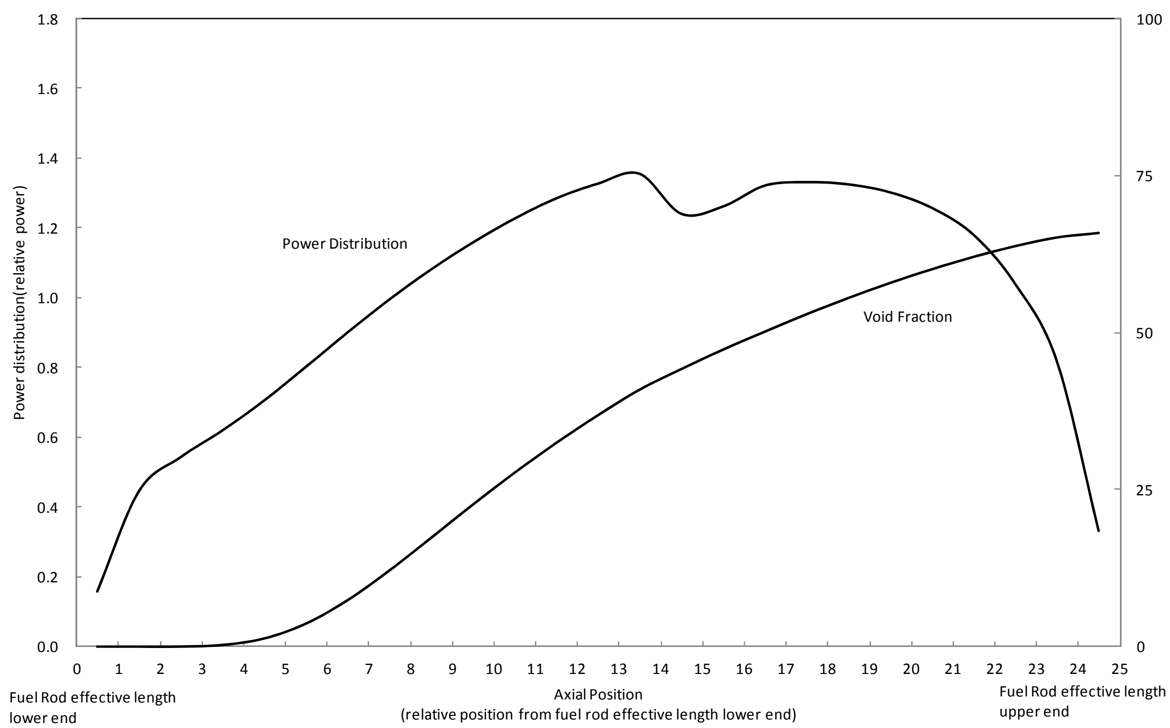


Figure 5.1.3.2-2: Axial Power and Void Fraction Distribution (End of Cycle)

5.1.4.Argument 1d: Detection and Management of Failed Fuel

The GE-14 nuclear fuel that will be used in the UK ABWR is designed, manufactured and managed to minimise the potential for fuel failures to occur that could subsequently result in the release of FPs into the steam circuit (5.1.1: Argument 1a: Design, Manufacture and Management of Fuel). In the unlikely event that a fuel failure occurs and FPs enter the steam circuit, the ABWR has a range of features (5.2.2: Argument 2b: Delay Beds for Noble Gases and Iodine) that allow for prompt detection and management of the failed fuel pin.

The gamma radiation level of the gases entering the charcoal adsorber in the Off-Gas System (OG) is continuously measured. The measurements are indicated and recorded in the Main Control Room (MCR). In addition, routine grab samples are spectroscopically analysed in the laboratory to obtain isotopic concentrations (5.1.4.1 Evidence: Detection System in Gaseous Waste Treatment System). At GDA Hitachi-GE considers it to be BAT to provide a radiation detector and grab sample analysis to enable the detection and management of a fuel failure (5.1.4.1 Evidence: Detection System in Gaseous Waste Treatment System).

The design of the UK ABWR reactor core allows the operator to detect the location of the failed fuel bundle by the selective insertion of control rods around fuel assemblies (5.1.4.2 Evidence: Procedures for Locating Fuel Failure in Reactor Core). Insertion of the control rods reduces nuclear fission in immediately adjacent fuel assemblies and allows the operator monitoring the concentration of FPs in gases from the condensers to detect the source of the leak. The source of any leak is identified when changes in the concentration of FPs entering the OG correlate with the movement of control rods at specific locations within the reactor core. Following detection of the failed fuel a future operator is then able to apply the failed fuel guidelines (e.g. continue an operation if the activity is less than the prescribed limit).

The design of the UK ABWR allows a graduated response to be taken to a fuel failure in the reactor and provides sufficient flexibility to allow a future operator to develop operating procedures to manage fuel failures associated with expected events and accident conditions.

At the next outage the failed fuel is removed and transferred to the SFP. In the SFP, a visual inspection is carried out to identify the failed rod and to determine the cause of failure. Other techniques including UT and inspection using a fibrescope are available to the operator to support the inspection if required. Depending on the severity of the failure the future operator has a number of options that can be utilised to store the fuel and to ensure that the spread of contamination is minimised. Not foreclosing options in terms of the storage of failed fuel and the selection of the optimal option by the future operator based on a risk assessment is the BAT (5.1.4.3 Evidence: Management of Failed Fuel).

5.1.4.1. Evidence: Detection System in Gaseous Waste Treatment System

The 'Approach to Sampling and Monitoring' report [Ref-20] describes the monitoring that is undertaken to measure the concentration of FPs entering the OG and to enable the detection of fuel failures and cladding defects. The monitoring arrangements include a gross gamma Continuous Emission Monitor (CEM) located at the inlet of the OG. The measurements are displayed and recorded in the MCR.

In the event of a fuel failure or cladding defect it is expected that radionuclides (such as Kr-83m, Kr-85, Kr-85m, Kr-87, Kr-88, Kr-89, Kr-90, Xe-131m, Xe-133, Xe-133m, Xe-135m, Xe-137, Xe-138, Xe-139) will be released directly from the failed fuel pin and ultimately enter the OG.

Hitachi-GE has demonstrated [Ref-20] that analysing a grab sample to identify specific radionuclides that are known to be released in the event of a failure will provide a future operator with the information required to manage such an event. The off-gas isotopic grab samples are taken periodically and analysed in the laboratory. In the event that an increase in radioactivity is detected, the frequency of collecting grab samples is increased to support the diagnoses and the management of the fuel failure. Procedures relating to the frequency of sampling will be produced by a future operator.

5.1.4.2. Evidence: Procedures for Locating Fuel Failure in Reactor Core

The UK ABWR has been designed to enable a future operator to identify and isolate a fuel bundle containing a fuel failure. When the abnormal increase of radioactivity is observed, the procedures are chosen based on the criterion of the activity level as follows:

1. **Raise the level of monitoring**. When the activity increases specifically and reaches a certain criterion (Criteria 1 in Figure 5.1.4.2-1), the operator increases the frequency of monitoring (e.g. increase the frequency of collecting a grab sample).
2. **Confirm if a fuel failure event has occurred**. When the activity increases to a certain criterion (Criteria 2 in Figure 5.1.4.2-1), the operator becomes aware that there is a fuel failure. The operator may conduct Power Suppression Testing (PST) to locate the failed fuel bundle as the activity level is sufficient to measure the response of the off-gas monitor during PST. A description of PST is provided in Figure 5.1.4.2-2.
3. **Plant shutdown**. When the activity increases up to a certain criterion (Criteria 3 in Figure 5.1.4.2-1) or continues to rise, the operator shuts down the plant.

Figure 5.1.4.2-1 also provides an overview of the procedures to be considered in the event of a fuel failure. Performance against the criterion is determined by analysis of grab samples (gas) and off-gas inlet radiation monitoring. The monitored activity depends on both the level of damage and the number of damaged fuel pins.

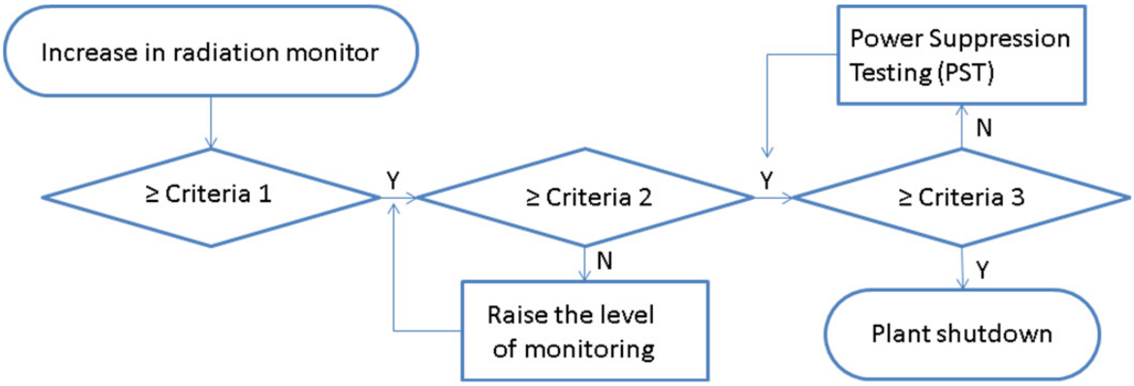


Figure 5.1.4.2-1: Operating Procedure against Fuel Failure

PST allows an operator to safely locate failed fuel in the core during operation and to subsequently enable the local management of the defect location to mitigate secondary damage and the associated fuel pellet material loss. This therefore helps to avoid mid-cycle outages following fuel failure.

The PST process consists of the following:

1) Transition to partial power with the insertion of control rods to prevent the damage of the failed fuel during PST.

2) Inserting and withdrawing one control rod or a certain group of control rods at a time. Failed fuel can be identified by the response of the OG monitor. Generally the fuel adjacent to the control rod with the largest activity change contains a failure.

3) Insertion of control rods around the suspected failed fuel channel to decrease its power and then keep the linear heat generation rate of its fuel rod below the limit.

4) Return to the rated power. Continue normal operations.

Figure 5.1.4.2-2: Description of PST

After the plant has been shutdown the concentration of FPs in the water inside each fuel assembly is measured to identify the failed fuel.

5.1.4.3. Evidence: Management of Failed Fuel

Once the plant has been shutdown, the failed fuel is moved from the core to the SFP using the same process used for all other fuel. Once located in the SFP the operator will undertake visual inspection to identify the failed rod and to determine the cause of the failure. Typically visual inspection is carried out using an underwater camera. If the failed rod or the cause of the failure cannot be identified using visual inspection the operator can use alternative techniques such as UT or using a fiberscope.

In accordance with relevant good practice [Ref-104] the option for storing failed fuel depends on the severity of the failure. For small pin hole failures such as those defined as an expected event the operator is likely to store the fuel directly in the SFP. The operator also has the option to isolate the fuel in a canister to prevent the spread of contamination if required.

A number of different options for the containerised storage of failed fuel rods are available, ranging from removal and storage of a single pin to the storage of a whole assembly in a vented filtered container. A range of options are also available to provide additional containment if the operator determines that it is appropriate to do so. These options are not foreclosed by the design of the UK ABWR and will be selected by the future operator based on a risk assessment performed on a case by case basis following identification of failed fuel.

Whilst the future operator will select the optimal storage method, the storage of failed fuel is not expected to result in the spread of contamination or an increase in the radioactivity of the SFP water. This is because the fuel will start to be cooled once it has been placed into the SFP.

5.1.5. Argument 1e: Commissioning, Start-up, Shutdown and Outage Procedures

Processes that have the potential to occur during commissioning, start-up, shutdown and outage of the reactor could result in an increase in direct doses to operators and a small increase in the generation of radioactive wastes. These are:

- An increase in the generation or mobilisation of Corrosion Products (CP) that are susceptible to activation as they pass through the reactor core.
- An increase in the incidence of SCC of key reactor components that will subsequently require replacement.

There are a number of techniques that have been developed for use during commissioning, start-up, shutdown and outage of the plant that collectively reduce the generation of radioactive waste during each operating cycle. These techniques have been successfully deployed on the (A)BWR reactor fleet in Japan.

Experience of operating BWRs in Japan has shown that as oxide films, which inevitably form on pipe and vessel internals, develop, they incorporate radionuclides, notably cobalt-60, resulting in a significant contribution to worker doses during outages. As such, during commissioning some pipes and vessels may undergo (as necessary) processes to create a film of oxide on internal surfaces (5.1.5.1 Evidence: Alkali Pre-Filming Technique). This ensures that the oxide layer is formed in a radiologically clean environment which prevents the later incorporation of radionuclides. This contributes to a reduction in the radiological dose from pipes in the Reactor Water Clean-up System (CUW) of approximately 80% as demonstrated in Japanese BWRs.

A negative consequence of this approach is that non-radioactive cobalt-59 ions will also not be accumulated on pipe and vessel internals. Cobalt-59 ions therefore have the potential to become activated as they pass through the reactor core. This will result in a very small increase in the concentration of radioactivity in spent demineralisers but it is deemed that the benefit from dose reduction outweighs the detriment associated with the small increase of radioactivity in the demineraliser resins.

Chemical decontamination is used to remove radioactivity from the internal surfaces of pipes and vessels if necessary. If a chemical clean is required the UK ABWR will likely use the series of Hydrazine, Oxalic Acid, Potassium Permanganate (HOP) methods developed by Hitachi-GE. This has been demonstrated in service to deliver Decontamination Factors (DF) of between 10 and 80 with a commensurate reduction in average dose rate of approximately 95 percent (5.1.5.2 Evidence: HOP Decontamination Process) and the wastes that are generated can be processed through the UK ABWR waste treatment systems. This minimises the quantity of secondary waste generated from decontamination processes.

Crud generated from ferrous materials that are present in the steam circuit after installation, maintenance or decontamination operations has the potential to pass through the reactor core and become activated. Plant start-up arrangements include recirculating condensate and feedwater through a condensate purification system prior to bringing the reactor back into service (5.1.5.3 Evidence: Water Conditioning). This allows any residual material in the condensate and feedwater systems to be captured in filters and minimises the potential to generate manganese-54 and iron-59 from the activation of ferrous materials which would otherwise then require treatment as radioactive waste.

Enhancements have been made to shutdown techniques by the adoption of Low Temperature Residual Heat Removal (RHR) Shutdown Cooling (5.1.5.4 Evidence: Low Temperature RHR Shutdown Cooling Method). This accelerates the cooling process and reduces the amount of radioactivity that is deposited on the internal surfaces of pipes by approximately 80 percent. The main benefit of this technique is that the radiological dose to workers is reduced during outage activities. However, the radioactivity of pipes will also be reduced with a commensurate reduction in the volume of maintenance related radioactive waste and decontamination fluids. Once the plant has been fully shutdown the condition of the water that remains in the pipes and vessels is carefully managed.

5.1.5.1. Evidence: Alkali Pre-Filming Technique

Oxide films develop on the surface of carbon steel pipes and components during normal operation of the NPP. If the oxide film is produced during reactor operation, cobalt-60 and other corrosion products entrained within the reactor coolant become deposited amongst the film and contribute towards worker dose during outages and increased ILW arisings. The alkali pre-filming technique intentionally produces an oxide film layer on the surface of carbon steel pipes in a non-radiological environment prior to reactor operation. This oxide film layer minimises the future formation of oxide films once reactor operation commences. This minimises corrosion products being deposited in the oxide film so contributes to reducing operator dose and ILW arisings.

Alkali pre-filming has the added benefit of being applicable to all carbon steel pipework and valves which have the potential for high radiological doses. Figure 5.1.5.1-1 provides an illustration of how alkali pre-filming prevents the deposition of cobalt-60.

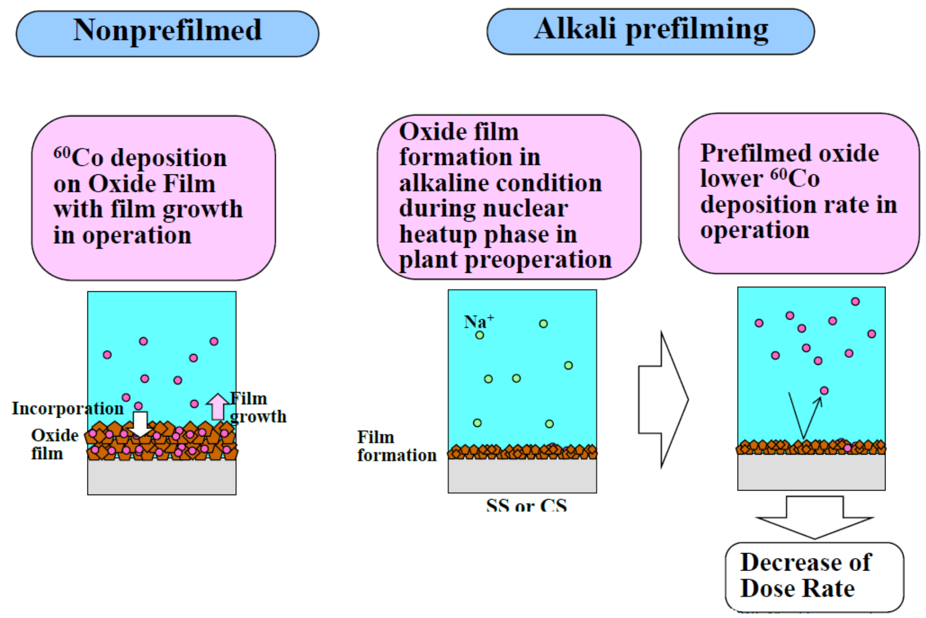


Figure 5.1.5.1-1: Illustration of the Alkali Pre-Filming Technique

The alkali pre-filming technique achieves a reduction in the deposition coefficient as shown in Figure 5.1.5.1-2, which in turn leads to a reduced dose rate, as shown in Figure 5.1.5.1-3 [Ref-63].

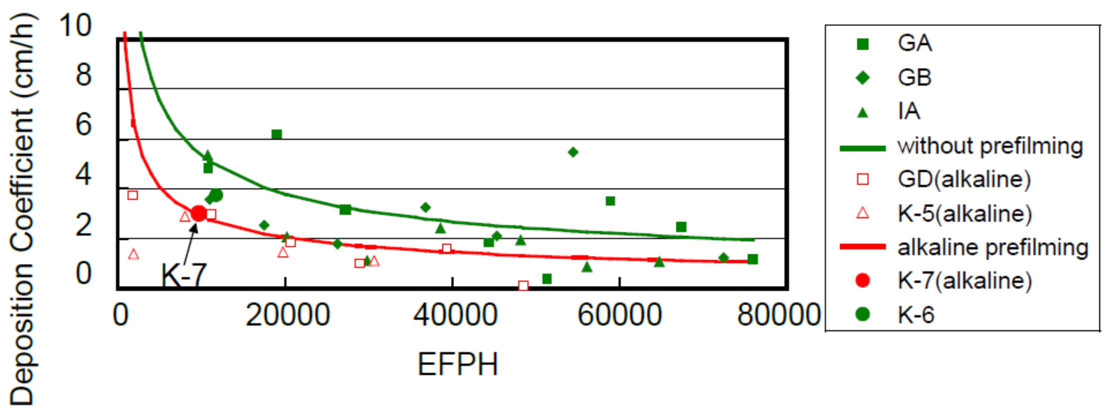
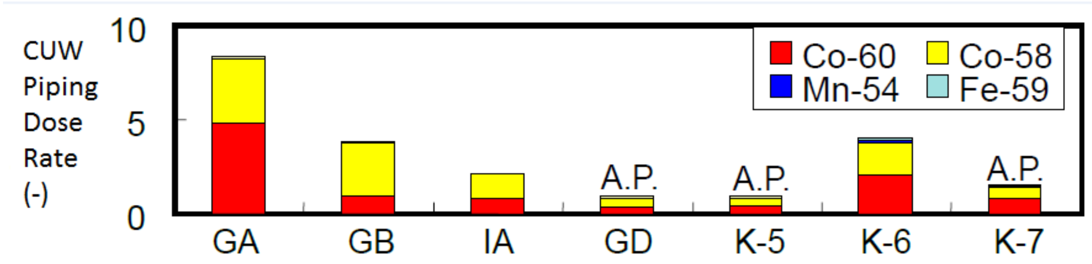


Figure 5.1.5.1-2: Cobalt-60 Deposition Coefficient on CUW Piping (Carbon Steel)



A.P. = NPP with alkali pre-filming

Figure 5.1.5.1-3: Dose Rate Comparison of Japanese NPP's with and without Alkali Pre-Filming (Carbon Steel)

5.1.5.2. Evidence: HOP Decontamination Process

HOP was developed by Hitachi and Kurita Engineering in order to clean pipe work by removing deposited radioactivity from internal surfaces. The chemicals of the HOP process are injected during outage resulting in a decontamination process that also removes any crud that is contained in the pipe work. If left in situ this crud could be transferred to the core, where it would become activated once the plant is operating and subsequently contribute towards the generation of radioactive waste.

The HOP process is undertaken at a temperature of 85-95°C and achieves a DF of between 10 and 80 [Ref-64].

The HOP process minimises the base metal corrosion of the pipes it comes into contact with, whilst the HOP chemicals are fully decomposed after use [Ref-64] [Ref-65].

5.1.5.3. Evidence: Water Conditioning

Water conditioning during start-up

Before start-up, the concentration of dissolved oxygen in the coolant is saturated due to exposure to air. High oxygen concentrations are undesirable as they contribute to SCC. To prevent this form of SCC, de-aeration is conducted and the dissolved oxygen concentration is controlled by maintaining oxygen concentration below 200 ppb before control rod withdrawal. The presence of high concentrations of oxygen also accelerates the corrosion of fuel cladding and as such, the de-aeration of the coolant contributes to maintaining the fuel integrity.

Any crud generated in the condensate system and feedwater system during an outage has the potential to be transported into the reactor at start-up, consequently resulting in an increase in radioactive CPs. To prevent activation of the crud, the feedwater and condensate system water is purified by re-circulation through the Condensate Filter (CF) to remove any crud prior to start-up. Further details of this process are provided in Section 5.1.8. Removing crud prior to it becoming activated reduces the radioactivity of the waste treatment systems (e.g. radioactivity of the demineraliser resins).

Water conditioning during shutdown

The radioactive crud concentration in the reactor coolant spikes during the decreasing pressure and temperature phases of shutdown. Crud may be deposited in areas of reduced flow, which results in the formation of radioactive 'hot spots'. The ABWR design minimises the formation of radioactive hot spots during shutdown by eliminating areas of stagnant flow or areas that may hold up material. Minimising the formation of radioactive hot spots during shutdown is effective at reducing operator radiological dose and also contributes to reducing the radioactivity of maintenance and decommissioning wastes[Ref-12].

Water conditioning during outage

During outage, the storage of the feedwater and condensate system will be optimised to minimise corrosion and the generation of Crud. Storage options include:

- Draining the system so that it is dry,
- Filling the system with deoxygenated water, or
- Filling the system with oxygenated water and recirculating it.

The storage method will be determined by the future operator and will depend on a number of factors, including the outage duration and the work taking place. Before start-up, the feedwater and condensate system is purified by re-circulation through the CF to remove any crud generated during the outage [Ref-12].

Management of Condensate Demineraliser (CD) resins

During an outage the CD is isolated and is stored in demineralised water to prevent degradation of the resin. By mitigating the deterioration of the resin the frequency at which the resin requires replacement is reduced and thus lower quantities of resin wastes are generated.

5.1.5.4. Evidence: Low Temperature RHR Shutdown Cooling Method

The Low Temperature RHR Shutdown Cooling Method is a lower RHR in-service temperature method compared to the former Soft Shutdown Method. The lower temperature of RHR in-service method reduces the amount of radioactivity that is deposited on the internal surfaces of pipes and vessels by approximately 80 percent [Ref-66]. Figure 5.1.5.4-1 outlines the process of the Low Temperature RHR Shutdown Cooling Method and Figure 5.1.5.4-2 shows the reduction in radioactivity deposition compared to the Soft Shutdown Method. The reduction in deposition of radioactivity on RHR piping and equipment results in a reduction in the radioactivity of maintenance related waste and decontamination fluids in addition to the reduction in direct dose to workers.

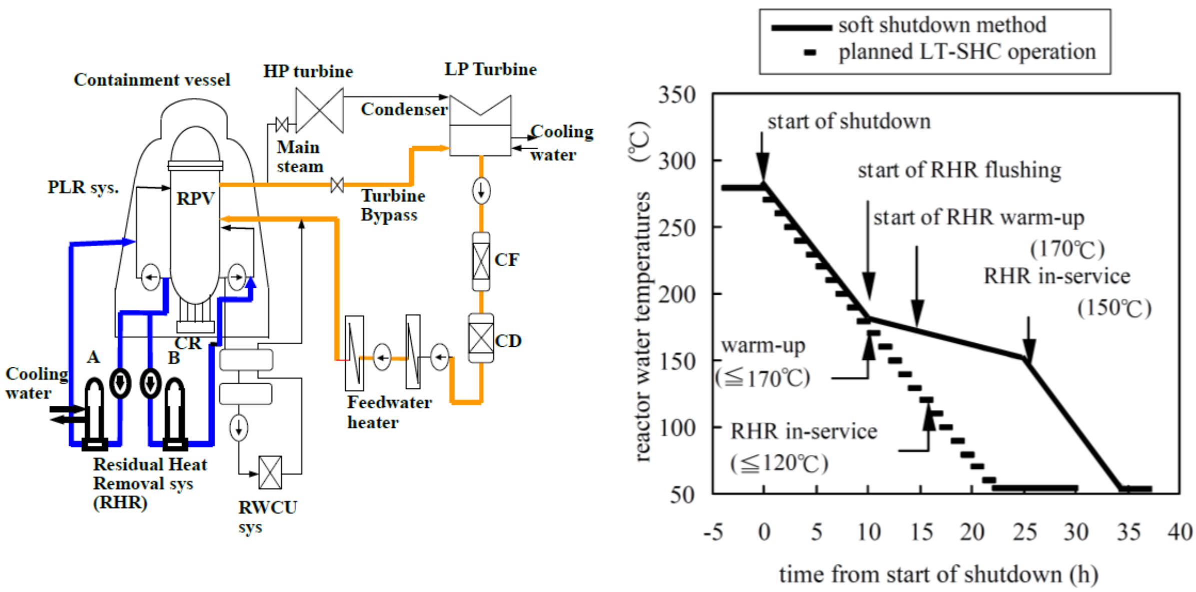


Figure 5.1.5.4-1: Low Temperature RHR Shutdown Cooling Method

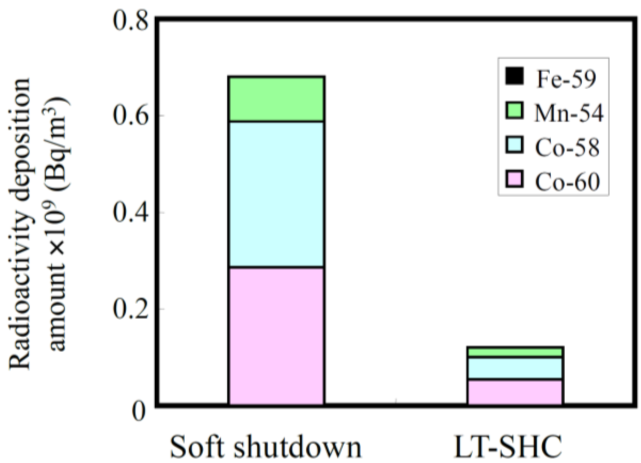


Figure 5.1.5.4-2: Comparison of Radioactivity Deposition between the Soft Shutdown Method (without LT-SHC) and Low Temperature RHR Shutdown Cooling Method

5.1.6.Argument 1f: Water Chemistry

The fundamental function of the water coolant in the UK ABWR steam circuit is to transfer heat from the fuel and to generate the steam necessary to drive the turbines. The chemistry of the coolant is carefully managed to deliver the following effects:

- reducing chemistry related failures of the fuel cladding (including corrosion and deposit related failures);
- minimising the generation of CPs that could become activated in the reactor core;
- minimising the generation of radioactive waste; and
- reducing occupational exposure of workers to ionising radiation.

The chemistry regime of the UK ABWR takes account of decades of optimisation of water chemistry in BWRs. The regime to be adopted in the UK ABWR will be hydrogen water chemistry with noble metal chemical addition, zinc addition, iron control and oxygen injection in the feedwater lines. These elements are summarised below:

- Hydrogen water chemistry is adopted to prevent the occurrence of Inter Granular Stress Corrosion Cracking (IGSCC) of SSCs. This reduces maintenance requirements and associated radioactive waste, reduces occupational exposure of workers and increases the availability of the plant (5.1.6.1 Evidence: Hydrogen Water Chemistry with Noble Metal Chemical Addition).
- Noble metal chemical addition (specifically On-Line NobelChem^{TM} (OLNC)) is employed to mitigate adverse effects of hydrogen water chemistry which lead to increased operator dose in other areas of the plant (5.1.6.1 Evidence: Hydrogen Water Chemistry with Noble Metal Chemical Addition).
- Zinc, in the form of depleted zinc oxide, is added to prevent cobalt-60 uptake on SSCs, made of stainless steel, which reduces operator doses and the release of cobalt-60 from fuel rod oxide layers which ultimately reduces the amount of cobalt-60 present in spent resins and filters (5.1.6.2 Evidence: Zinc Injection).
- The concentration of iron present in the feedwater is reduced and controlled to limit the amount of corrosion product containing iron oxide film that can form on the fuel and to ensure that what is formed does not readily dissolve/spallate. This reduces the release of corrosion products, most notably cobalt-60 so reduces the activity in spent demineraliser resins (5.1.6.3 Evidence: Iron Concentration Control).
- Oxygen is injected into the feedwater to prevent corrosion and reduce the concentration of iron cruds in the reactor water (5.1.6.4 Evidence: Oxygen Injection).
- Additionally, progressive improvements have been made to the water coolant clean-up system including the provision of a condensate clean-up system and a CUW. The two systems work together to minimise detrimental effects on the fuel performance, to reduce the generation of activation products and associated radioactive waste and to reduce occupational exposure to workers (5.1.6.5 Evidence: Condensate Clean-up System and Reactor Clean-up System to Remove Corrosion Products).

The management of water chemistry has been a fundamental element undertaken by Hitachi-GE to reduce the failure of fuel, to reduce operator and public dose, to minimise the generation of radioactive waste and to improve plant availability. Data available for BWRs show that by the early 1990s the number of fuel failures had reduced to approximately 10 percent of the peak experienced in 1974 and that availability of the plant continued to show an upward trend during this period (5.1.6.6 Evidence: Fuel integrity).

5.1.6.1.Evidence: Hydrogen Water Chemistry with Noble Metal Chemical Addition

Early generations of BWRs employed a Normal Water Chemistry (NWC) regime which did not use any chemical dosing agents. The primary objective of this regime was to control the conductivity and chloride levels to prevent the occurrence of Trans granular Stress Corrosion Cracking (TGSCC) in austenitic stainless steels. The NWC approach was effective at mitigating TGSCC but resulted in the presence of oxygen and hydrogen peroxide in the reactor water by radiolysis. This created an oxidising environment within the reactor coolant water which led to the occurrence of IGSCC.

As a result, HWC was introduced in the 1980's. This involved the injection of hydrogen into the feedwater as a way of reducing IGSCC [Ref-96]. With HWC, the reactor water becomes a reducing environment where oxidising species such as oxygen and hydrogen peroxide are consumed by the recombination reaction with hydrogen. The effect of injecting hydrogen into the feedwater is to reduce the Electrochemical Corrosion Potential (ECP) and hence decrease crack growth rates. Whilst residual stress remains in the material surface, reduction of the corrosion potential has the effect of greatly reducing crack growth. Laboratory tests found that the addition of hydrogen to the feedwater to reduce the ECP to less than -230mV (relative to that of the standard hydrogen electrode (SHE)) was optimum in suppressing SCC.

A disadvantage of the approach was that it also changed the chemical form of nitrogen-16, formed in the reactor coolant water by neutron activation of oxygen-16, from soluble nitrate compounds to non-soluble forms such as nitrogen oxides and ammonia. This caused the nitrogen-16 to partition into the steam phase. Although the half-life of nitrogen-16 is very short and does not impact on discharges to the environment it does result in an increase in gamma shine within the turbine building (T/B) during normal operations and therefore an increased dose to workers and the public.

To mitigate the undesirable volatilisation of nitrogen-16, Noble Metal Chemical Addition (NMCA) was therefore employed alongside HWC. This involved the addition of the noble metals such as platinum and rhodium which catalyse the combination of injected hydrogen with oxygen and hydrogen peroxide. This reduced the amount of hydrogen which needed to be injected (from about 2ppm to 0.15-0.35ppm [Ref-96]) into the feedwater to achieve the desired combination of hydrogen with oxygen and H_2O_2 which in turn reduced the amount of Nitrogen-16 which was volatised and transferred over tclassio the steam phase.

Classic NMCA involved the addition of platinum in the form of $Na_2Pt(OH)_6$ and rhodium in the form of $Na_3Rh(NO_2)_6$ with the application being performed during outages at a frequency of approximately once every 4 to 5 years. However, $Na_3Rh(NO_2)_6$ was found to contain impurities, specifically chlorine, which becomes activated and leads to the undesirable generation of radioactive chlorine-36.

As such, OLNC was later developed and represents an optimised approach for achieving the desired catalysation. OLNC allows the application to be performed at power at a frequency of once every year. OLNC employs only platinum (in the form of $Na_2Pt(OH)_6$) so eliminates the chlorine-36 production associated with the use of $Na_3Rh(NO_2)_6$. This is additionally beneficial from a corrosion perspective, as chloride is also stated as a control parameter in the Water Quality Specification [Ref-86] (and EPRI water chemistry guidelines) due to its aggressive nature and influence on SCC.

Addition of $Na_2Pt(OH)_6$ does introduce sodium which becomes activated to form radioactive sodium-24. Due to sodium being non-reactive in this environment, it builds up in the reactor coolant water and the spent demineraliser resins [Ref-97]. However, this is largely non-volatile so does not readily partition to the steam phase and has a half-life of only 15 hours so is not persistent in solid radioactive wastes. As such, the benefit of adding the platinum catalyst is considered to outweigh the detriment of introducing of sodium.

The benefit of NMCA is illustrated in Figure 5.1.6.1-1 which shows the main steam line dose rates on an ABWR as a function of feedwater hydrogen concentration highlighting the typical hydrogen concentrations and associated dose rates for the three regimes (NWC, HWC and HWC + NMCA) [Ref-97].

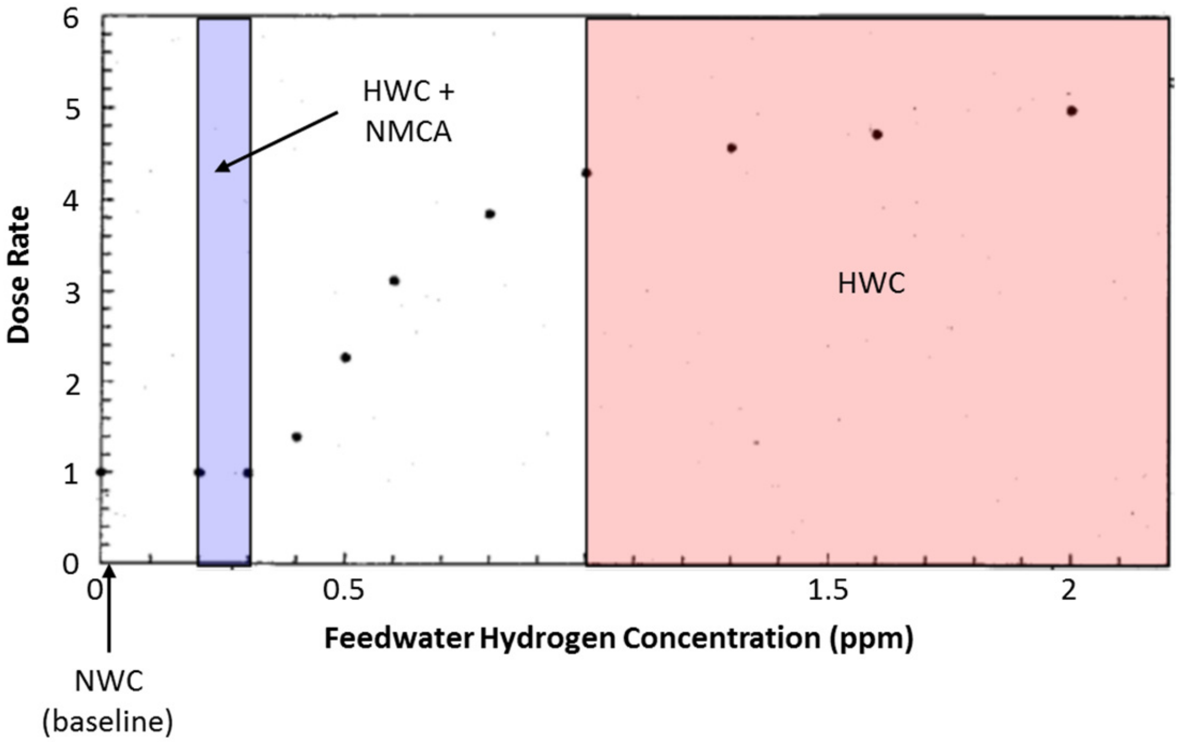


Figure 5.1.6.1-1: Relation between feedwater hydrogen concentration and main steam dose rate at Kashiwazaki-Kariwa-7

5.1.6.2. Evidence: Zinc Injection

Zinc injection is currently used in BWRs for the reduction of pipe work dose rates. Zinc is injected continuously from the feedwater system [Ref-12]. Injecting zinc into the reactor circuit suppresses the build-up of cobalt-60 inside the oxide films that form on the surface of pipes and vessels which would result in increased direct radiological doses to operators [Ref-12]. It is reported that the impact of zinc was first recognised by the observation that NPPs with high zinc concentrations in reactor water had lower radiation dose rates from pipe work. Through experimental and field studies, it was found that elevated concentrations of soluble zinc in reactor water reduced cobalt-60 build-up in the corrosion films on pipe work and components by promoting the formation of a protective oxide film. Radiation dose rates are lowered since zinc is favoured for incorporation into the oxide film relative to cobalt-60. While Natural Zinc Oxide (NZO) is effective at reducing operator doses, activation of the zinc-64 (48 percent of NZO isotopic composition) to form zinc-65 contributes to an increase in operator dose and the generation of radioactive waste. Consequently, Depleted Zinc Oxide (DZO), that is, zinc depleted in zinc-64, is commonly used for zinc injection. When DZO is coupled with both optimum iron control and cobalt source control, low shutdown operator dose rates are achieved.

A qualification programme for the use of zinc addition to reduce shutdown dose levels also indicated a further positive effect on the susceptibility of components and pipe work to Intergranular Stress Corrosion Cracking (IGSCC). Research concluded that low levels of zinc reduced crack propagation rates under reducing conditions, and that high concentrations of zinc reduced crack growth rates even under oxidising conditions. Beneficial effects of zinc on crack initiation and growth of Alloy 600 were also observed in PWR environments. Therefore, work was initiated to find a specification to optimise the zinc concentration and hydrogen injection rate that would minimise operator dose rates during normal operations whilst providing IGSCC mitigation for the bottom region of the reactor vessel.

Investigations have also been conducted to ensure there are no adverse effects from zinc addition on fuel cladding surface crud deposition. For example, zinc deposits can form a tenacious, insoluble spinel form of zinc ferrite on the fuel cladding, and there has been concern with respect to spalling and corrosion from these crud deposits. Therefore, monitoring the effect of adding chemicals to the coolant has been a focus of recent fuel surveillance programmes. However, no gross adverse effects of hydrogen, zinc and NMCA, when added within prescribed amounts, have been observed. No gross hydriding has been identified either, although detection methods are not capable of detecting minor hydriding.

5.1.6.3. Evidence: Iron Concentration Control

During normal operation, corrosion products (such as iron, nickel, chromium, zinc and cobalt) are released from structural materials and are present in the reactor coolant water with iron representing the major component of these corrosion products [Ref-97].

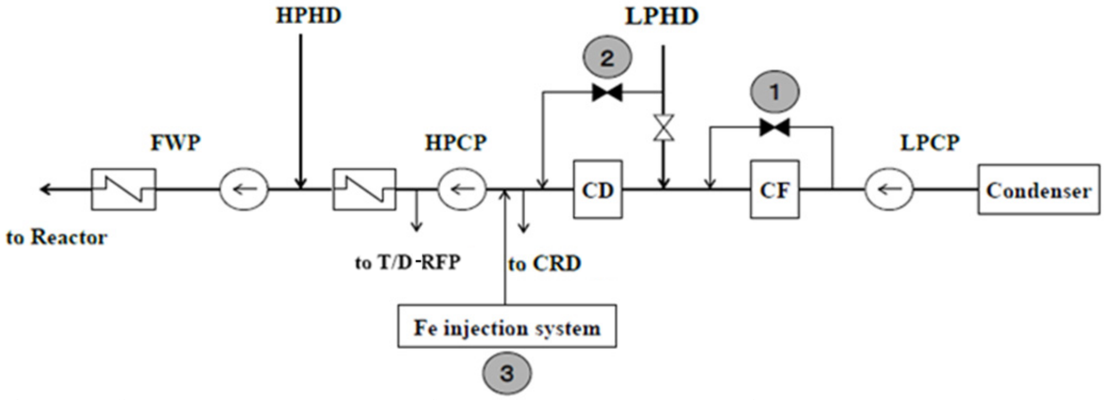
The iron crud then deposits on the surface of the fuel elements as an oxide layer. As this oxide layer forms it also incorporates other dissolved corrosion products (such as nickel, chromium, zinc and cobalt) and these become part of the matrix of the oxide layer. Since this oxide layer is subjected to the intense neutron flux in the core, the incorporated corrosion products become activated giving rise to radioactive activation products. The oxide layer containing the activation products can then undergo spallation and, albeit limited, dissolution which then releases the activation products into the reactor coolant water as both particulates and soluble ions [Ref-97].

Limiting the amount of iron present in the feedwater of the reactor limits the amount of iron available to form the oxide layer on the fuel surface. This in turn reduces the amount of dissolved corrosion products that become activated and released into the reactor coolant water [Ref-97].

Additionally, another benefit of limiting the feedwater iron concentration is that it ensures that any oxide layer produced (oxide layer formation can be reduced but cannot be eliminated) is less susceptible to dissolution and spallation. Oxide layer fuel deposits actually consist of two layers; a tenacious inner layer enriched in transition metals and an outer non-adherent layer. Reducing the feedwater iron concentration reduces the size of the outer layer, whilst maintaining the inner layer, thus preventing the release of activity [Ref-97]. This is important as it ensures that the activation product containing oxide layer stays attached to the fuel (and is removed with the fuel elements during refuelling) and does not readily become released into the reactor coolant water.

Conversely, there is OPEX to show that in some plants, if the iron concentration is reduced too low soluble monoxides such as NiO, CoO, ZnO may form on the fuel cladding surface instead. However, these species readily dissolve back into the reactor water resulting in increased reactor coolant water activation product concentrations [Ref-97].

The UK ABWR is expected to be a low iron plant. The approach for the UK ABWR is therefore to control the iron concentration in the feedwater to <1.0 ppb and let it trend towards its natural level. If an increase in cobalt-60 is observed HNP have the option to then operate under optimum iron control (iron concentration of 0.1-1.0ppb) using the condensate filter (CF) bypass [Ref-97]. Procedures for sample collection, testing and analytical reporting will specify the utilisation of the CF bypass line from grab sampling and analysis of iron. Action levels will be set on iron concentration to determine when to actuate the bypass valve for control over the volume of water bypassing the CF.



Item	①	②	③
Method	CF Bypass	LPHD mixed in CD Downstream	Fe injection into Feedwater
Feature	-Easy to operate after stabilizing of CD iron removal efficiency -No additional injection system -Increase in back-wash frequency of CD	-Effective for increasing small amount of Fe -Rigid -Sufficient amount of Fe can not be injected -Other corrosion products such as Co and Ni are fed into the reactor	-Flexible, Easy -Short term, Auxiliary -Additional injection system is required -Necessity of operation and maintenance of injection system
Range of Fe conc. control	0.3-1 ppb	0.4-0.5 ppb	0.1-1 ppb

Figure 5.1.6.3-1: Iron Concentration Control Methods

5.1.6.4. Evidence: Oxygen Injection

Since general corrosion correlates with Flow Accelerated Corrosion (FAC), the prevention of both crud formation and FAC of carbon steel in the feedwater and condensate system depends on the same countermeasures. It is known that a concentration of dissolved oxygen of more than 15 ppb can minimise the occurrence of both types of corrosion [Ref-12]. The dissolved oxygen concentration in the feedwater and condensate system is therefore maintained at or above 15ppb to prevent corrosion [Ref-85] by application of a Water Chemistry Control procedure that instructs on the monitoring of oxygen and sets an action level for oxygen injection.

Figure 5.1.6.4-1 shows conductivity, iron concentration and dissolved oxygen concentration in the feedwater line when oxygen injection is implemented [Ref-11]. The oxygen concentration is measured at the inlet and outlet of the CF, outlet of the CD and outlet of the feedwater heater. The oxygen injection point is set between the CF and the CD and the concentration of dissolved oxygen ranges from 20 to 30 ppb in feedwater with oxygen injection [Ref-85]. The concentration of iron crud and conductivity decreases with oxygen injection, which indicates that the corrosion of carbon steel is reduced.

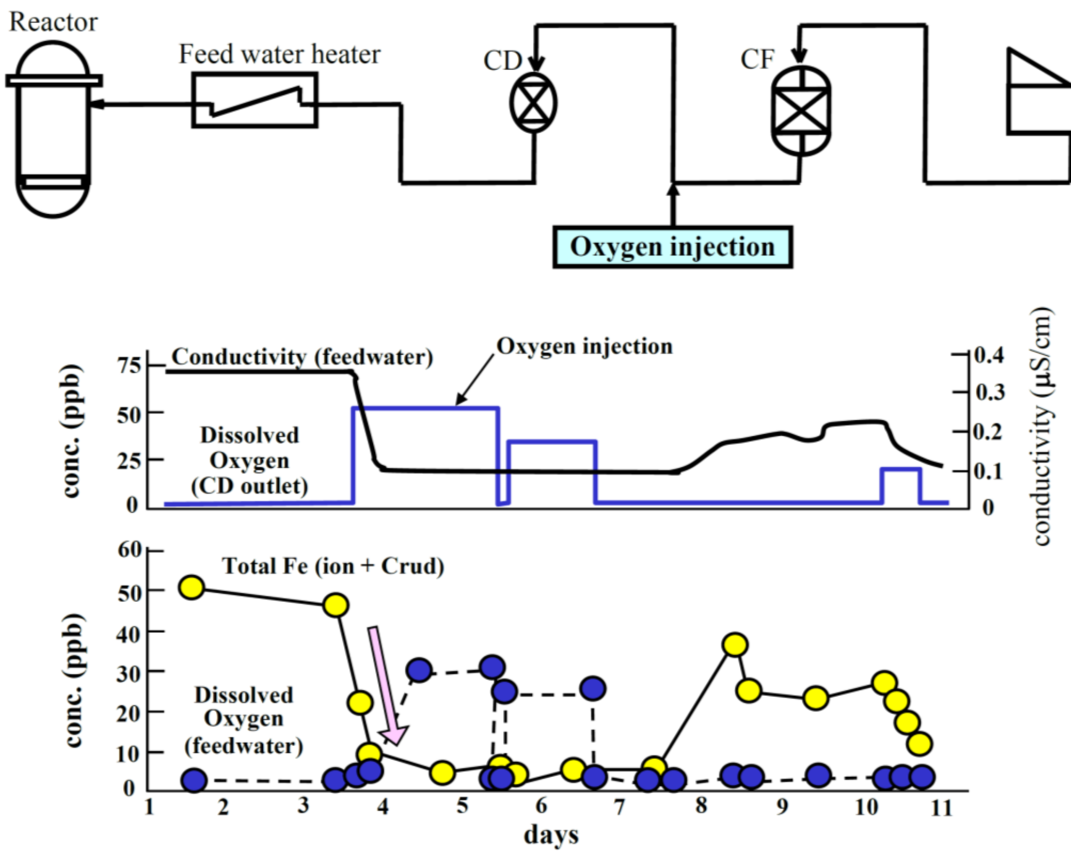


Figure 5.1.6.4-1: Conductivity, Iron Concentration and Dissolved Oxygen Concentration of Feedwater Line with Oxygen Injection

5.1.6.5. Evidence: Condensate Clean-up System and Reactor Clean-up System to Remove Corrosion Products

The amount of CPs in the feedwater and condensate system is controlled to minimise detrimental effects on the fuel performance, operator dose and the generation of radioactive waste. Increased concentrations of CPs within the core region has the potential to effect material integrity and fuel structural material integrity by accumulating on the fuel cladding and thereby decreasing thermal conductivity. By reducing the amount of CPs there are less activation products which results in a reduction in direct doses to operators and less radioactive waste being generated.

To prevent the incorporation of impurities, CFs and CDs are installed on the Condensate Clean-up System and Filter Demineraliser (FD) on the CUW. A description of these systems has been provided in 5.1.8 Argument 1h: Recycling of Water to Prevent Discharges.

FDs also help to reduce the occurrence of SCC as they reduce the concentration of chloride ions and sulphate ions which increase the SCC susceptibility of materials.

5.1.6.6. Evidence: Fuel integrity

The number of fuel failure events has declined since the late 1970s as illustrated in Figure 5.1.6.6-1 [Ref-11]. This is a result of fuel cladding material improvements (5.1.1 Argument 1a: Design, Manufacture and Management of Fuel) and water chemistry controls. Water chemistry measures introduced to maintain the integrity of the fuel include the purification of coolant and the introduction of measures that reduce the generation of CPs. The deposition of harmful impurities such as chloride and CPs on fuel surface (fuel deposits) strongly affects the integrity of fuel claddings. Fuel deposits have been reduced through methods such as oxygen injection (which is described in more detail in section 5.1.6.4 Evidence: Oxygen Injection).

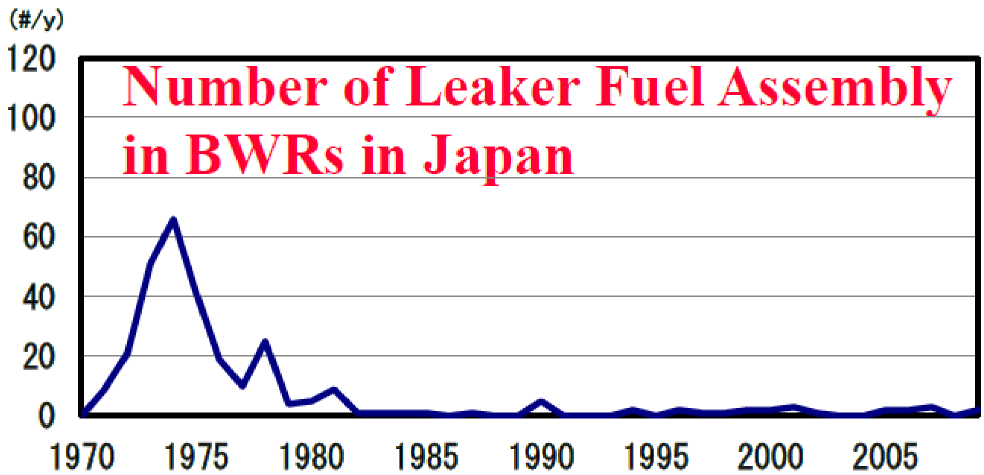


Figure 5.1.6.6-1: Reduction in Number of Fuel Leaks Follow Material and Water Chemistry Improvements

5.1.7. Argument 1g: Specification of Materials

Materials in the reactor are exposed to neutrons generated by nuclear fission. In some instances the materials will become radioactive by a process known as 'activation'. These 'activation products' are a significant source of direct doses to workers and are a source of radioactive waste. There are two main sources of activation products:

- Structural materials within the reactor that are activated by their proximity to nuclear fuel and the associated neutron flux. These materials become radioactive waste during maintenance and decommissioning tasks.
- CPs that are suspended in the reactor water deposit on the surface of the fuel cladding as a result of boiling densification and become activated. The activated elements can then be re-dissolved into the reactor water and subsequently have the potential to contribute to an increase in dose.

The UK ABWR takes account of decades of experience in the design, operation and decommissioning of LWRs and, where practicable, uses materials that are less susceptible to corrosion, deposition and activation. Efforts to use alternative materials have sought to balance the benefits provided by the characteristics of the materials with their safety and environmental implications.

Cobalt is present as an impurity in the stainless steels and nickel based alloys that are used within the UK ABWR reactor and parts of the steam circuit. Cobalt is also a significant component of cobalt based alloys (e.g. Stellite®) that have been historically used in reactor systems due to their hard-facing properties. The naturally occurring isotope cobalt-59 becomes activated by neutrons to create the radioactive isotope cobalt-60. Progressive evolutions of the BWR design have sought to reduce the amount of cobalt present in materials of construction. The design of the UK ABWR now includes the specification of Low Cobalt Material (LCM) for a range of components that are particularly susceptible to direct activation or from which corrosion products may become activated (5.1.7.1 Evidence: Specification of Low Cobalt Materials). The design also limits the use of Stellite® to those components where it is essential for material performance (5.1.7.2 Evidence: Substitution of Stellites®).

CPs generated in the condensate system have been reduced by the progressive introduction of corrosion resistant steels over several evolutions of the (A)BWR design. The CPs contain the naturally occurring isotopes iron-56 and iron-58 which, when activated, becomes iron-59 or manganese-54 (5.1.7.3 Evidence: Introduction of Low Corrosion Materials). The CPs generated also contain nickel and chromium which, when activated, become cobalt-58 and chromium-51.

In combination, these improvements have resulted in an expected 50 percent reduction in the amount of cobalt-60 generated during the operation of the ABWR compared to if these improvements were not applied (5.1.7.4. Evidence: Specification of Materials – Operational Experience and Feedback).

These design elements all contribute to eliminating or reducing the generation of radioactive waste by either removing readily activated isotopes from materials used or preventing materials from becoming mobile and migrating into or through areas of high neutron flux where they can become activated.

5.1.7.1. Evidence: Specification of Low Cobalt Materials

Improvements in material specifications have aimed to eliminate the use of cobalt based alloys wherever possible and reduce cobalt in stainless steels and Ni-based alloy that are used for large surface area or high flow rate components. LCM is used to reduce activated radioactivity and cobalt-60 concentrations in the reactor water. Reducing the activated radioactivity of components results in a reduction in activity of those components when they are disposed of as waste during maintenance and decommissioning. Preventing the generation and release of cobalt-60 into the reactor water minimises the generation of radioactive waste when the reactor water is treated by the CUW.

A materials specification will be developed to limit the cobalt content of stainless steel, Ni-based alloys and welding materials. This specification will apply to those materials which are particularly susceptible to activation or dissolution with subsequent activation. A design review and subsequent ALARP assessment was performed to assess the use of LCM [Ref-50]. The design review explored opportunities to increase the use of LCM in the design of the UK ABWR. The review evaluated the cost of applying LCM, the potential reduction in direct dose to workers and the generation of radioactive waste for a range of reactor components. It was concluded that it was both ALARP and BAT to manufacture the components in the reactor from LCM (≦0.05 percent cobalt) with some exceptions which only make a small contribution to activation and cobalt release.

The materials used in the reactor and steam system consist mainly of austenitic stainless steel, nickel based alloys, carbon steel and low alloy steel components. The cobalt content of the alloys used in the core region (control rods, fuel bundle, steam separator and dryer and some internal reactor components), as well as final feedwater heater tubes, is minimised to reduce the creation of activation products and associated potential for gamma radiation. To ensure that the concentration of cobalt is minimised a material specification has been developed for the components of these systems that require the cobalt content to be less than 0.05 percent [Ref-13].

It is recognised that the use of LCM in specific components reduces both direct doses to operators and the generation of radioactive waste. Hitachi-GE has therefore undertaken an assessment of the availability of material with a cobalt content that is less than the current limit of 0.05 percent proposed to be used in the UK ABWR. The assessment, based on Japanese suppliers concluded that the limited availability and additional cost of reducing the cobalt content limit to lower than 0.05 percent was grossly disproportionate compared to the benefits [Ref-50]. A preliminary review of European suppliers confirmed the findings of the assessment. However, it is considered proportionate that a future operator should undertake a detailed assessment of availability and costs within the UK context (FA1010).

5.1.7.2. Evidence: Substitution of Stellites®

Cobalt based alloys are almost completely eliminated inside the Reactor Pressure Vessel (RPV) of the ABWR. This has been achieved by:

- Eliminating jet pumps found in conventional BWRs, which had Stellite® present in the slip joints.
- The use of iron and nickel based alloys for control rod rollers and pins rather than cobalt based alloys.

There is also the potential to replace valve seats manufactured from cobalt based alloys that are located external to the reactor vessel with alternative materials.

Stellites® are cobalt based alloys with hard-facing characteristics. Stellite® surfaces exposed to reactor coolant will contribute to cobalt in feedwater which will become activated in the reactor core producing cobalt-60. In BWR designs, Co-based alloys such as Stellites® are used in taps, valves and some parts of the internal core support structure and primary pumps [Ref-13].

In the UK ABWR design, Co-based alloys will be replaced by a cobalt-free material with confirmed sufficient material characteristics, or, in some cases, by an adequate nickel and iron base. In previous designs, cobalt content in Stellite® was between 45 to 64 percent. However, under the current design of the UK ABWR the cobalt content is less than or equal to 1.0 percent as a result of the selection of alternative materials [Ref-13]. Examples of material substitution are:

- the use of Wear Proof Material (WPM), a nickel based alloy, for rollers used for the control rod; and
- Nitronic®-60 that is an iron based alloy used for control rod pins.

The cobalt content of both alternate materials is 0.25 percent or less.

5.1.7.3. Evidence: Introduction of Low Corrosion Materials

To reduce iron input into the reactor water, extensive use of low alloy steel and stainless steel in pipe work systems and other components will be implemented. The effect of adopting corrosion resistant steel, along with the introduction of a dual condensate polishing system and oxygen injection, has reduced the iron feedwater concentration from between 5 to 10 ppb to less than 1 ppb (5.1.6 Argument 1f: Water Chemistry). This has reduced the generation of iron-59 and manganese-54 on the fuel cladding and thus reduced the generation of radioactive waste and dose rates.

Other significant corrosion products are nickel and chromium which become activated to form cobalt-58 and chromium-51. Cobalt-58 radiologically exhibits almost the same behaviour as cobalt-60. Chromium-51 has a short half-life and low gamma energy which means it is less significant in terms of radioactive waste generation and radiation exposure.

5.1.7.4. Evidence: Specification of Materials - Operational Experience and Feedback

The improvements of using stainless steels and nickel based alloys (Inconel®) with reduced concentrations of cobalt have led to a reduction in the amount of cobalt-60 activated in the core. This is illustrated in Figure 5.1.7.4-1 [Ref-13].

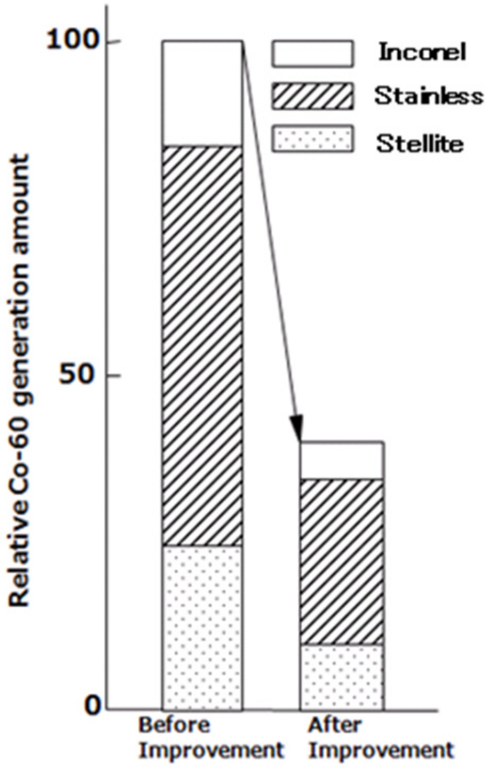


Figure 5.1.7.4-1: Reduction of the Generation of Co-60 as a Result of Using LCM and Reducing Stellite®

5.1.8.Argument 1h: Recycling of Water to Prevent Discharges

Water is used as the coolant within the UK ABWR and to generate the steam that is used in the turbines to generate electricity. The water coolant becomes contaminated with radioactive material as it passes through the reactor and around the steam circuit. A high concentration of radioactivity in the steam circuit is undesirable as it can result in increased operational exposure to workers during operational and maintenance activities. The disposal of the contaminated water is also undesirable as it could harm members of the public and the environment.

The design of the UK ABWR includes the following systems for recycling water that is used during operations and maintenance:

- The Condensate Water Clean-up System treats water that has passed through the turbines and has been condensed back to water. Filters remove any solid matter that could either damage the fuel or become activated in the reactor core. Demineralisers are used to remove ions that have become activated or have the potential to become activated in the reactor core or deposit on internal surfaces of pipes and vessels (5.1.8.1. Evidence: Condensate Water Clean-up System).
- The CUW continuously draws water from the reactor and passes it through a FD system to control conductivity and remove impurities. This has the combined effect of reducing the potential for corrosion of fuel cladding, minimising the generation of CPs on internal surfaces and reducing the potential for CPs to become activated in the reactor core. Once treated, the water is returned to the reactor in the main feed line. The quality of the water passing through the system is continuously monitored to ensure that the characteristics are within defined parameters and that the system is performing as expected. In the unlikely event that the characteristics of the liquid fall outside of the defined parameters, the FD performance can be regenerated by performing the backwash operation. The water is then recirculated and passed through the cleaned FD so that the

characteristics of the liquid meet the defined parameters. During start-up and shutdown operations, any excess reactor water can be transferred to the LCW to enable an operator to manage the reactor water level of the system. Following treatment, this water is then returned to the CST and is then available to be recycled back into the reactor water circuit (5.1.8.2. Evidence: Reactor Water Clean-up System).

- The Fuel Pool Cooling and Clean-up System (FPC) and the Suppression Pool Clean-up System (SPCU) use shared demineralisers to maintain the water quality. The characteristics of the water in these two areas is broadly similar as water from the UK ABWRs Suppression Pool (S/P) is pumped to the equipment lay-down pool during refuelling operations. At the end of refuelling, water from the reactor well is returned to the S/P where its quality continues to be managed by the SPCU. This represents an improvement on previous generations of the BWR which used water from the CST during refuelling operations which was subsequently transferred to the liquid effluent system and then back to the CST. The quantity of water previously extracted from the CST per refuelling operation was approximately 2500 m^3 in the BWR-5. Eliminating this process has resulted in reducing the size of the CST and the inventory of water in the reactor circuit (5.1.8.3 Evidence: Fuel Pool Cooling/Clean-up System and the Suppression Pool Clean-up System).
- The LCW treats liquid waste from the SFP, FPC and CUW, along with liquid waste from equipment drains, etc. The LCW consists of a filter and a demineraliser which ensures that the water quality meets the criteria for the CST and subsequent reuse in the plant (5.1.8.4. Evidence: LCW Treatment System).
- The TGS uses water extracted from the CST to produce steam that is used in the turbine gland seal. Following use in the turbine gland, 98 percent of the steam is condensed and is subsequently returned to the main condenser and is made available for reuse (5.1.8.7 Evidence: Recycling of Water with Steam Circuit – Operational Experience and Feedback).

The design of the water treatment systems described above includes demineralisers to remove soluble material including radionuclides. Demineraliser (also referred to as ion-exchange) systems are utilised throughout the nuclear industry (5.1.8.5 Evidence: Nuclear Industry Application - Demineralisers) to remove soluble material including radionuclides from liquid processes to maintain the water quality within target values.

The demineralisers are capable of using a variety of resins (5.1.8.6 Evidence: Demineraliser Media) which provides sufficient flexibility to allow a future operator to select the resin based on operating requirements and compatibility with subsequent disposability requirements. This flexibility is considered to represent BAT at GDA. The future operator will determine the selection of demineraliser resins (FA2).

Recycling water throughout the UK ABWR mitigates the requirement to make liquid discharges from the steam circuit during the operational life of the facility (5.1.8.7 Evidence: Recycling of Water with Steam Circuit – Operational Experience and Feedback). Liquid effluent is also reused during decommissioning activities to prevent the generation of new liquid waste (5.1.8.8 Evidence: Reuse of liquid effluent during decommissioning activities). A disposal of liquid effluent from the steam circuit will only take place when the facility is decommissioned. The performance of these systems will be confirmed during commissioning (FA1).

5.1.8.1. Evidence: Condensate Water Clean-up System

The configuration of the Condensate Water Clean-up System is illustrated in Figure 5.1.8.1-1. CFs remove any crud in the system which could either damage the fuel or become activated in the reactor core. Demineralisers are used to remove ions that have become activated or have the potential to become activated in the reactor core or become deposited on internal surfaces of pipes and vessels.

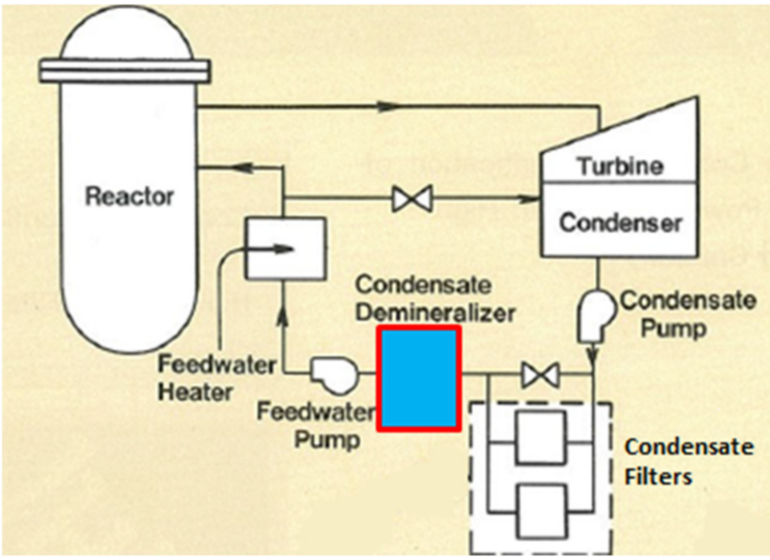


Figure 5.1.8.1-1: Condensate Water Clean-up System

Design improvements led to the CD being used in combination with a pre-filter known as the CF [Ref-13]. Prior to the introduction of the CF any crud in the water remained in the system and contributed to increased operator dose and an increase in the radioactivity of maintenance wastes. The pre-filter removes insoluble ions and the crud content resulting in the following improvements:

- Operator dose reduction;
- Reduction in radioactivity of maintenance wastes; and
- Prevention of the demineraliser blocking as a result of solids build up.

The performance of the CF is summarised in Table 5.1.8.1-1.

Table 5.1.8.1-1: Impact of Introducing Demineraliser Pre-filters

	Before (system before improvement)	**After (Improved & standardised system)**
Requirement for Condensate Polishing System	Ion removal	Ion removal & crud removal for reduced radiation exposure around reactor
Outlet metallic impurities	Approx. 10ppb	<Approx. 1ppb
Occupational Exposure	Approx. 10person·Sv	<Approx. 1person·Sv*1

Note) *1: Condensate Pre-Filter is one of the improvements of plant dose rate reduction.

The CD is an external non-regenerative type (demineralising by mixed bed ion exchange resins) and processes approximately 900 m^3/h of water from the condenser [Ref-9]. The demineraliser removes soluble salts (substances present in ionic form) from the condensate allowing the radioactivity to be disposed of as solid waste once the resins are spent. The CDs will be non-regenerative, but will require frequent “washing”, to remove deposited Crud.

5.1.8.2. Evidence: Reactor Water Clean-up System

To maintain the parameters of the reactor water and to minimise the transfer of impurities into the steam circuit a CUW is provided. The CUW system conditions the coolant using filter demineralisers to remove impurities in order to reduce the potential for corrosion of fuel cladding, minimise the generation of CPs on internal surfaces and reduce the potential for CPs to become activated in the reactor core. The configuration of the CUW is illustrated in Figure 5.1.8.2-1.

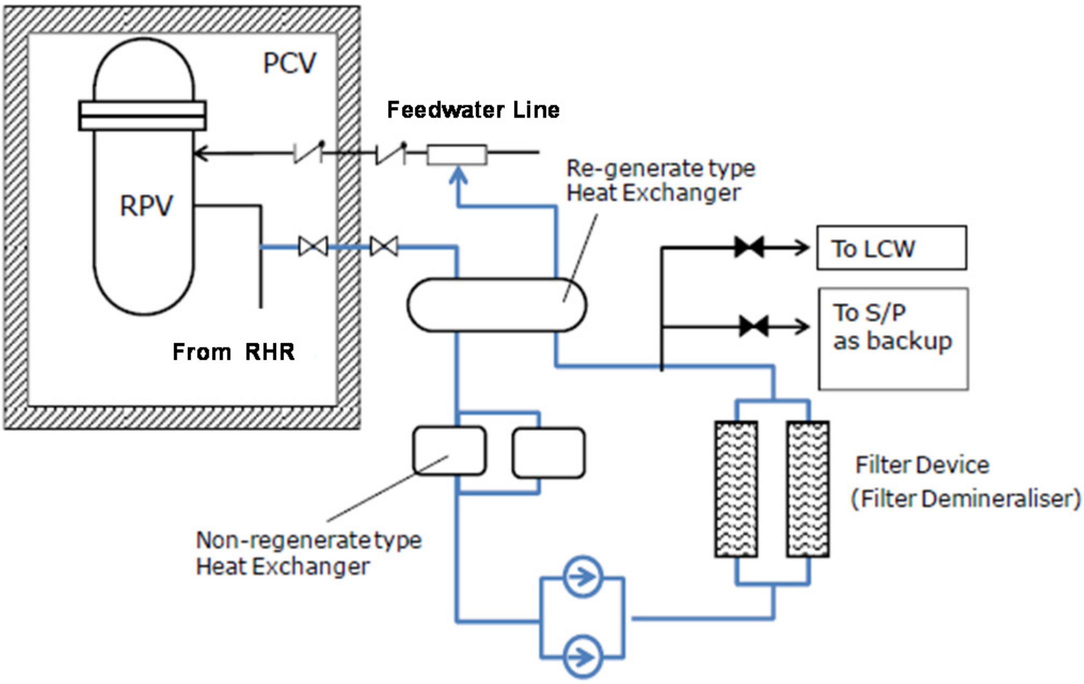


Figure 5.1.8.2-1: CUW

The CUW removes impurities contained in the reactor water and maintains the quality of the reactor water within the determined range in order to prevent:

- Corrosion of the equipment and pipe work of the reactor cooling system;
- Decreasing the heat transfer efficiency by avoiding the adhesion of impurities on the fuel surface; and
- Radioactive contamination of the reactor cooling system and the related equipment.

The CUW system, through the use of filter demineralisers, seeks to maintain the coolant within the design specifications presented in Table 5.1.8.2-1 [Ref-86].

Table 5.1.8.2-1: Design Specification Coolant Conditions Following Treatment in the CUW

Conductivity (25°C)	100 µS/m or less
Cl^-	100 ppb or less
pH (25°C)	5.6 to 8.6

In practice the system performs much better than the conditions in Table 5.1.8.2-1 and can achieve 10 µS/m at 25°C. In the unlikely event that the parameters of the coolant fall outside of the design specification prescribed in Table 5.1.8.2-1 the operator can review the efficacy of the treatment system. If required the filter demineralisers can be backwashed and the coolant can be recirculated through the filters and demineralisers so that the characteristics of the liquid meet the defined parameters. During start-up and shutdown operations, any excess reactor water can be transferred to the LCW to enable an operator to manage the reactor water level of the system. Details of the treatment in the LCW are described in Section

5.2.5 (Argument 2f: Configuration of Liquid Management Systems). Following treatment in the LCW the water is returned to the CST Tank where it is reused.

5.1.8.3. Evidence: Fuel Pool Cooling/Clean-up System and the Suppression Pool Clean-up System

The FPC provides two functions:

- removal of decay heat from the SF stored in the SFP; and
- remove impurities from the water of the fuel pool.

Figure 5.1.8.3-1 shows the configuration of the FPC which includes heat exchangers for the removal of decay heat and filter demineralisers for the removal of impurities. The filter demineralisers are designed to process approximately 250m³/h of water [Ref-9].

The purpose of water chemistry management of the SFP is primarily to maintain the fuel integrity in the storage pool, including the integrity of the storage rack and pool itself and to minimise the radioactivity level increase in the pool. For this purpose, coolant is cleaned up and its temperature is controlled during normal operations.

Coolant purification has two purposes:

- the first is to protect the fuel, fuel rack and pool structure from corrosion; and
- the second is to maintain the clarity of coolant to allow visual inspection of the fuel assembly located under water.

In the initial stages of BWR development, a large amount of activated crud was dispersed inside the pool during refuelling resulting in higher operator doses. However, in recent years, because of the reduction of CPs from the feedwater line, the problem of elevated radiation levels during refuelling has been dramatically improved. The water chemistry of the SFP in operation is sufficiently controlled such that no failure caused by water chemistry related damage has been experienced in ABWR designs.

Further detail on the FPC and the role that it plays within the whole SFP is provided in the Pre Construction Safety Report (PCSR) Chapter 19: Fuel Storage and Handling [Ref-88].

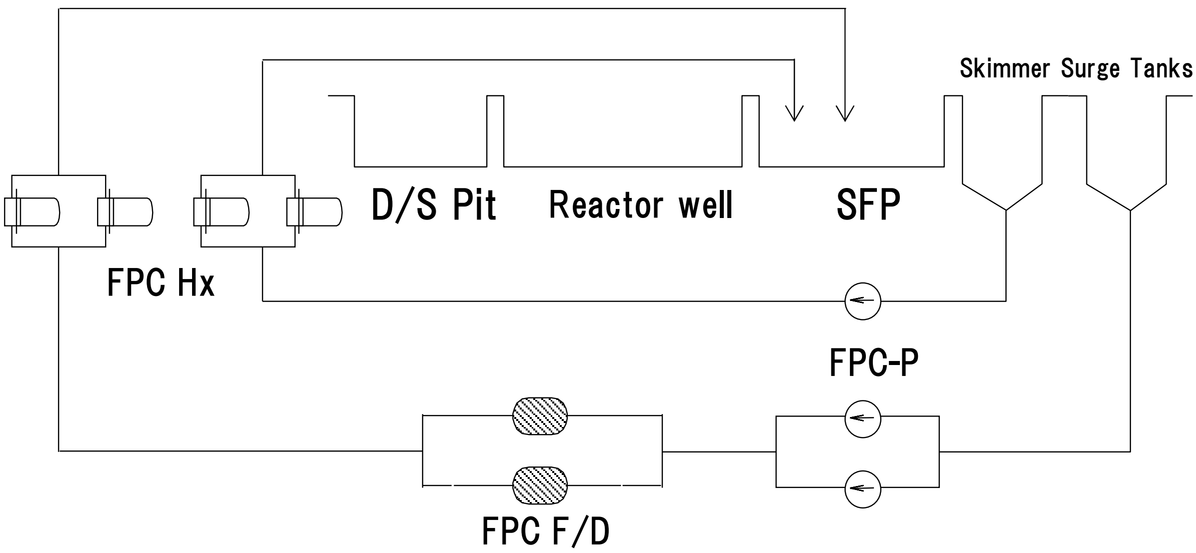


Figure 5.1.8.3-1: FPC

Figure 5.1.8.3-2 shows the SPCU system configuration. This system shares the same filter demineraliser as the FPC which removes impurities from the pool water in the Suppression Chamber (S/C).

As well as maintaining the water quality of the S/P, the SPCU also has the function of transferring water to the upper pools (Equipment Lay Down Pool and Reactor Well) prior to refuelling. See Figure 5.1.8.3-3. This represents an improvement to the previous BWR-5 design which requires the upper pools to be filled with water from the CST. By adopting the SPCU and utilising the SP water, a decrease in demand on the radioactive waste treatment system and the CST has been achieved, allowing for its capacity to be reduced by 40 percent [Ref-9]. Figure 5.1.8.3-4 summarises this design improvement.

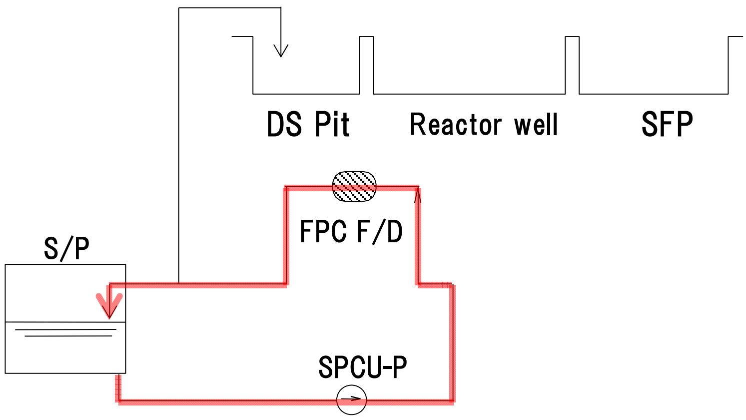


Figure 5.1.8.3-2: SPCU (S/P Water Clean up Mode)

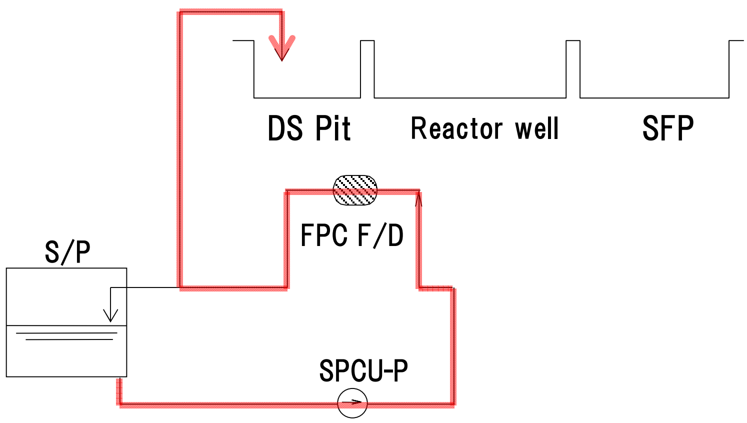


Figure 5.1.8.3-3: SPCU (Water Filling Operating Mode)

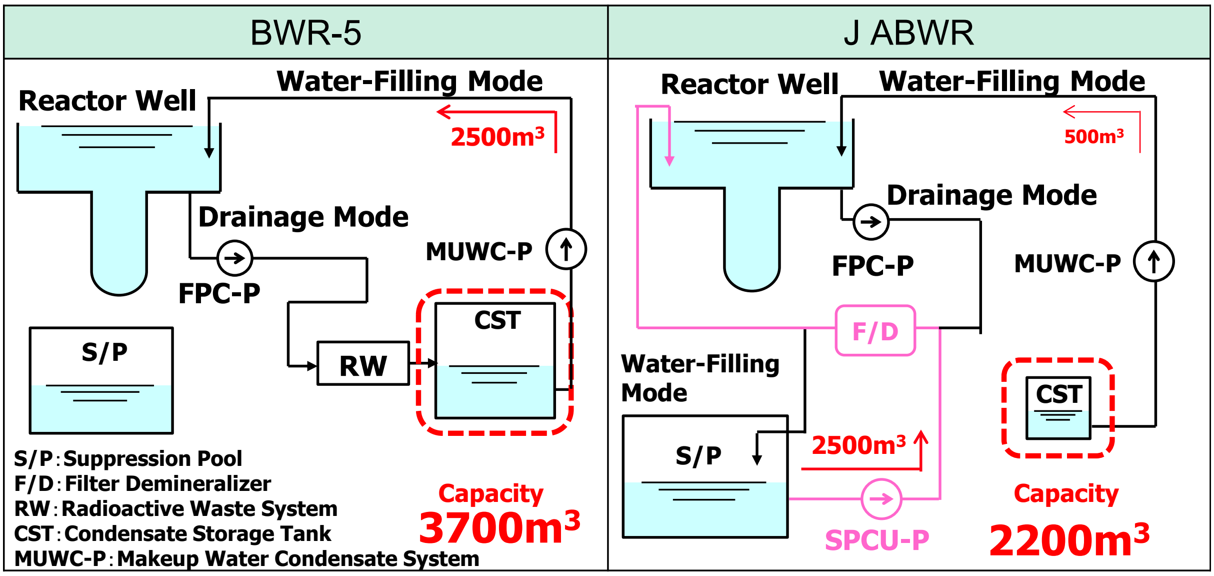


Figure 5.1.8.3-4: Design Improvement of Adopting the SPCU [Ref-9]

5.1.8.4. Evidence: LCW Treatment System

LCW is liquid waste from the SFP, FPC and CUW, along with liquid waste from equipment drains, etc. LCW from the SFP, FPC and CUW is excess liquid that is removed from these systems during reactor start-up and shutdown operations to help maintain the water balance of the plant. The LCW treats the liquids which are then returned to the CST for reuse.

The LCW consists of filters, for the removal of insolubles, demineralisers, for the removal of solubles, and sampling pools, as shown by Figure 5.1.8.4-1. The LCW therefore allows LCW to be treated and reused rather than being discharged during normal operation. Further detail on the LCW is provided in PCSR Chapter 18: Radioactive Waste Management [Ref-58].

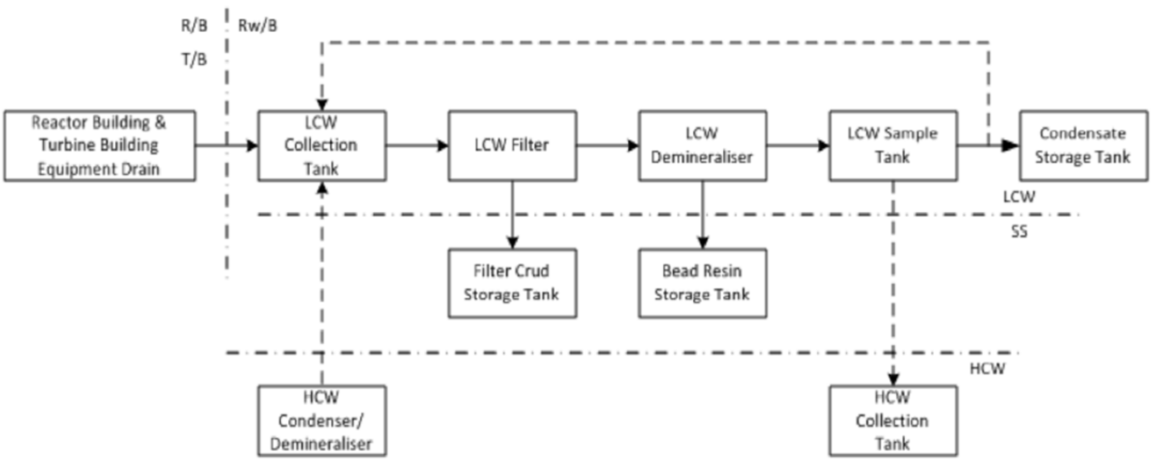


Figure 5.1.8.4-1: Schematic of the Configuration of the LCW

Hitachi-GE undertook an assessment [Ref-59] to compare the different treatment technologies available for the LCW. The assessment compared demineralisers (ion exchange), reverse osmosis membrane and cross-flow filtration against a range of criteria such as operational experience, reliability, maintainability, solid waste generation, DF and cost. Demineraliser technology scored significantly higher than the other two techniques, outperforming them in the following areas:

- lowest impact on the generation of solid waste;
- lowest capital cost;
- highest reliability safety;
- highest maintainability;
- highest resistance to radiation; and
- low/no requirement for chemicals.

The assessment therefore identified the use of demineralisers, as well as a filter, as the preferred option for treatment of LCW.

At the time of this initial assessment, the assessment team did not have access to detailed OPEX from Japanese operational facilities. Their assessment was therefore based on the technologies, which were judged to be best practice at Japanese operational sites and for which there was also UK OPEX. Upon the availability of additional detailed OPEX, Hitachi-GE undertook a further options assessment [Ref-107] to investigate all credible liquid waste options, to identify good practice and to ensure the chosen options are underpinned by robust and transparent justifications.

This subsequent assessment [Ref-107] concluded that the baseline option (demineralisers and filters) meets the minimum requirements for the system, achieves the requirements of good practice and has significant successful OPEX to support its use. Additionally, it noted that the other options assessed provide no further safety or environmental benefit over the baseline option. It was therefore decided that filters and demineralisers were the preferred option. This supports the conclusion of the initial assessment [Ref-59].

The filtration system will remove up to 99 percent of the insoluble contamination and suspended solids from the effluent. The ion exchange step will remove a similar proportion of the soluble component of the radioactive contamination. As a result, the average expected radioactivity within the effluent will be reduced to levels lower than the current reuse criteria of <37 Bq/cm^3.

The performance of the LCW that is achieved in standard ABWRs has been demonstrated through operational experience and feedback. Analysis of samples taken from the inlet and outlet of the LCW demonstrates a DF of approximately 1,000. This high level of performance is required to ensure that liquids are suitable to be reused within the NPP.

5.1.8.5. Evidence: Nuclear Industry Application – Demineralisers

Demineralisation (also known as ion exchange) removes soluble materials from aqueous streams. The process of demineralisation is the removal of soluble salts (substances present in ionic form) from aqueous effluents using ion exchange resins which retain certain substances which are then converted into solid waste when the demineralisation medium is spent. Ion exchange resins are recognised as RGP in the nuclear industry for the removal of soluble radionuclides and are used extensively on NPPs in the UK and internationally. In its GDA Public Consultation Document on the UK EPR, the Environment Agency stated that “at this time, filtration by cartridge filter, ion exchange and, for effluents incompatible with ion exchange, evaporation are BAT for use in the UK EPR”. It is therefore concluded that for similar wastes the use of demineralisation is also BAT for the UK ABWR.

5.1.8.6. Evidence: Demineraliser Media

Ion exchange media selection depends on the properties of the target ion, the presence of other competing ions in the feed stream, availability and cost. The capacity of the media relates to how much target ion a particular type of media can hold.

Ion exchange media typically used in demineralisers in UK nuclear installations are either made of:

- organic resins, which can carry various functional groups that provide a cation or anion exchange effect; or

- inorganic ion exchangers, some of which act as adsorbers rather than ion exchangers and, to make them more efficient, are fabricated into beads or microporous gels with a high surface area.

The media used in the Japanese ABWR is consistent with the types listed above and is expected to be the same as those to be used in the UK ABWR.

The choice of demineralisation media represents a balance between three factors:

- **The overall type of wastes to be treated**. Some waste streams have a specific and consistent composition which allows a particular demineralisation media to be used at maximum efficiency. However, in other cases the composition of the waste stream and its variability require a more general-purpose demineralisation media to be used.
- **The nature of the nuclide to be removed**. If a specific nuclide is to be removed from the waste stream, a specialised demineralisation media can be chosen. However, if a more general clean-up is required a media that offers an overall good performance needs to be used (e.g. incorporating anion and cation beds).
- **The quantities of secondary waste arisings**. A demineralisation media might offer good selectivity for a particular radionuclide or waste stream but may have a short lifetime, so resulting in higher secondary solid waste arisings.

The Reactor Water Clean-up System (CUW), the FPC, SPCU and LCW all use a combination of filters and demineralisers to treat the water to allow reuse. The design of the demineralisers on these systems is sufficiently flexible to allow a future operator to select the most appropriate ion exchange media. Ion exchange media are consumables and the type of media used in the treatment systems might be adapted throughout the operational life of the plant in order to ensure that BAT continues to be applied during the life time of the NPP. The flexibility provided by the demineralisers is limited to the modification of the ion-exchange resin used based on:

(i) Cation and anion ratio changes which is linked to the water quality to be treated; and
(ii) The selection of low- Total Organic Carbon (TOC)-elution ion-exchange resin, high-spec-ion-exchange resin, etc.

This flexibility will allow a future operator to alter the performance of the demineralisers, operational life time of the resins and to further optimise treatment of the water. This flexibility is considered to represent BAT at GDA.

5.1.8.7. Evidence: Recycling of Water within Steam Circuit - Operational Experience and Feedback

The treatment systems described in sections 5.1.8.1 to 5.1.8.4 allow for the water of the steam circuit, the SFP and the S/C to be recycled and thus alleviates the need to make liquid discharges from these systems under normal operating conditions.

The TGS design uses water extracted from the CST to produce steam that is subsequently used in the turbine gland seal. The use of CST water as the supply for the gland steam evaporator rather than purified water allows the operator to manage the water balance of the plant without having to make additional discharges of aqueous radioactive waste.

Figure 5.1.8.7-1 shows the water balance of the plant. The majority of the water is recycled within each of the treatment systems described above however, in some cases, water may be sent to the LCW for further treatment. This is then recycled back to the CST for reuse. There are therefore no routine discharges from these systems. The only liquid discharges from the UK ABWR are from the Laundry Drain (LD), Controlled Area Drain System (CAD) and occasionally the HCW, as described in 5.2.5 Argument 2f: Configuration of Liquid Management Systems.

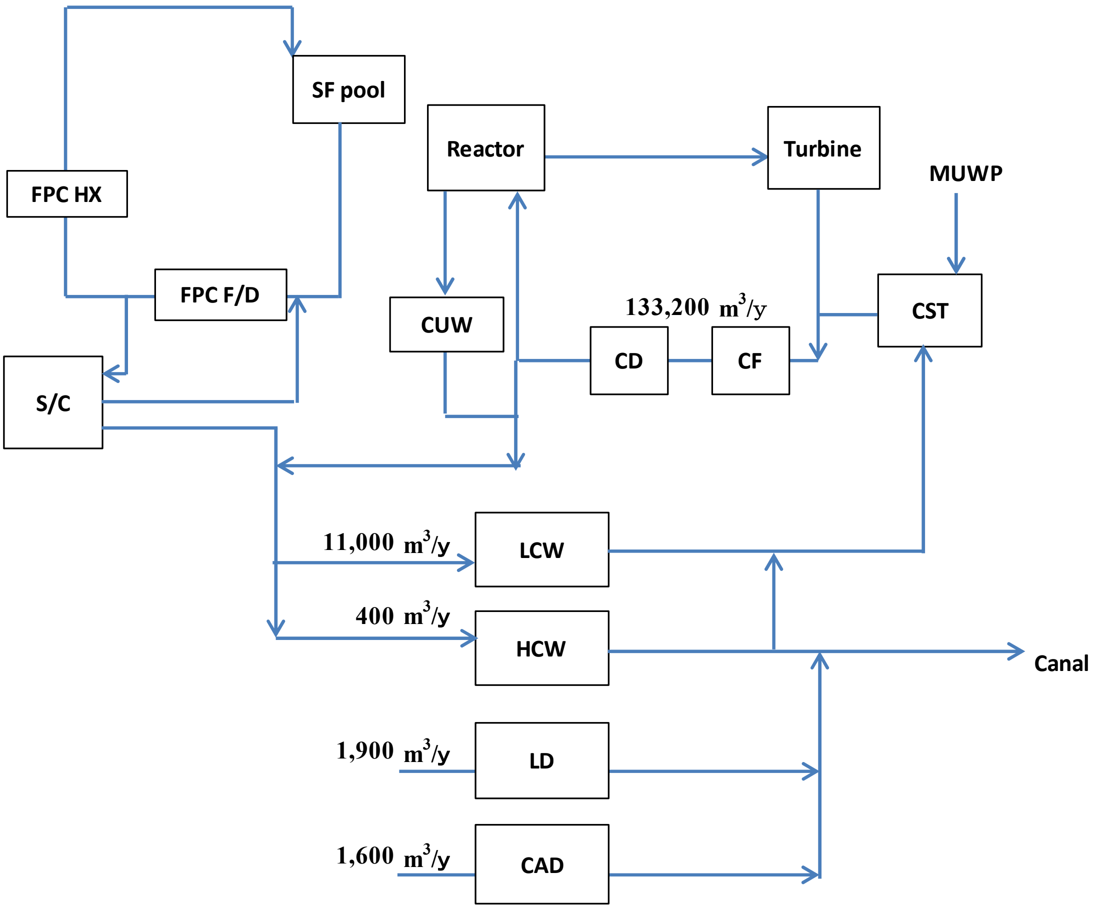


Figure 5.1.8.7-1: Water Balance of the Plant Showing the Recycling of Water

5.1.8.8. Evidence: Reuse of liquid effluent during decommissioning activities

Water is required during decommissioning of the UK ABWR for processes such as chemical decontamination and underwater cutting of RIN. Rather than introducing clean water from outside of the system to carry out these processes, which would then become contaminated and lead to increased volumes of radioactive discharges, a proportion of the plant water that has been retained within the NPP during the 60 years of operation could be used. This reused plant water has relatively low levels of radioactivity due to decay over the plant lifetime whilst its retention and reuse during decommissioning activities reduces the activity further. Reusing the plant water saves the generation of 7,870m^3 of new liquid effluent that would have been required for decommissioning activities. Of the 11,710m^3 of water retained after shut down of the plant, around 67 percent could be reused as shown in Table 5.1.8.8-1. Ultimately the future operator will determine whether to reuse plant water during decommissioning activities [Ref-100].

Table 5.1.8.8-1: Reuse of plant water for decommissioning activities

No.	Decommissioning activity	Required water volume during decommissioning (m^3)	Retained water (m^3)	Reused rate (%)
1	Decontamination of systems	750		
2	RIN dismantling	3,310		
3	RPV dismantling	2,160	-	-
4	Reactor shield wall (RSW) dismantling	1,350		
5	High pressure water jet washing	300		
Total		7,870	11,710	67

Prior to eventual discharge to the environment the liquid effluent is treated by the existing RW/B equipment or other temporary liquid waste treatment facility. The discharges are not expected to exceed the operational discharge limits proposed by Hitachi-GE [Ref-10].

5.1.9.Argument 1i: Secondary Neutron Sources

Secondary neutron sources provide additional neutrons, at a controlled rate, to assist with reactor start-up. PWRs typically use antimony-beryllium neutron sources which generate tritium. The cladding for antimony-beryllium neutron sources is typically manufactured from stainless steel which is porous to tritium. Any tritium that is generated can therefore diffuse through the cladding and into the reactor coolant. The UK ABWR design will use californium-252 as the start-up neutron source, instead of antimony-beryllium. An advantage of using californium-252 over antimony-beryllium neutron sources is that it does not promote the generation of tritium. The ABWR design also means that secondary neutron sources are only required during the start-up phase of the first fuel cycle. Due to the flexibility provided by the reactor design (as described in 5.1.3: Argument 1c: Efficiency of Fuel Use), whereby fuel with a range of burn-ups can be used, secondary neutron sources are unlikely to be required after the first cycle and can therefore be removed.

The selection of materials for secondary neutron sources is limited. There are very few materials that have the combined features of neutron generation and a sufficiently long half-life to make them suitable for operations in a nuclear reactor. Neutron sources typically fall into one of three types which are spontaneous fission sources, alpha-neutron sources and photoneutron sources. The selection of the source material will have largest impact on the amount of tritium that enters the water coolant. The reference case for the UK ABWR is that californium-252 sources in stainless-steel cladding will continue to be used as there is considerable operational experience available and that the quantity of tritium generated is relatively low.

5.1.9.1.Evidence: Selection of Neutron Source Materials

The neutron source assemblies consist of two main components – a long cylindrical tube assembly, referred to as the source holder, which is supported by the core plate and the top guide. The assembly is installed and removed in the same manner as a local power range monitor (LPRM) instrument tube, with a spring-loaded upper tip that fits under the top guide. Each source holder houses a neutron source pin assembly containing the californium-252 neutron source material.

96.9 percent of californium-252 (2.6 years half-life) undergoes alpha decay to generate curium-248 ($3.4x10^5$ years half-life) while the remaining 3.1 percent of decays are spontaneous fission. Use of californium-252 eliminates the production of tritium. Tritium is generated from antimony beryllium as a result of the following reaction chain:

$$
\begin{aligned}
&{}^{9}_{4}Be+{}^{1}_{0}n\rightarrow{}^{4}_{2}\alpha+{}^{6}_{2}He\\
&{}^{6}_{2}He\xrightarrow{t_{1/2}=807ms}{}^{0}_{-1}\beta^{-}+{}^{6}_{3}Li\\
&{}^{6}_{3}Li+{}^{1}_{0}n\rightarrow{}^{4}_{2}\alpha+{}^{3}_{1}H
\end{aligned}
$$

5.1.9.2. Evidence: Selection of Neutron Source Cladding

The californium-252 is in the form of palladium- californium composite wire. The palladium-californium wires are contained in welded stainless steel capsules (inner capsule) of Type 316L stainless steel. The inner capsule in turn is inside a 316L outer capsule, which provides a second encapsulation and forms the neutron source pin assembly, exposed to the reactor coolant. The wire consists of a uniform distribution of californium in a palladium alloy matrix. The palladium matrix itself constitutes a form of encapsulation.

The compatibility of the palladium-californium wire with 316L stainless steel capsule is established. Palladium has a relatively high melting point (1552°C) and is resistant to degradation by oxidation and corrosion. Regarding chemical compatibility, no reactions have been found between the palladium-californium wire and stainless steel up to 1000°C.

5.1.9.3. Evidence: Secondary Neutron Sources - Operational Experience and Feedback

The californium-252 neutron source was commercially available from the 1970s and it was first used for the start-up of a commercial reactor in 1973. Today, the use of californium-252 neutron sources for start-up is applied for all new reactors in Japan and the U.S.

From the 1980s Hitachi-GE has applied the californium-252 neutron sources to the start-up of eight reactors. Californium-252 neutron sources are unloaded 1-2 cycles after start-up.

These sources have also been used for reactors that have experienced an extended shutdown period of 7 to 10 years. In this case, the irradiated fuel alone may not have provided an adequate source of neutrons.

5.1.10. Argument 1j: Leak Tightness of Liquid, Gas and Mixed Phase Systems

The design of the UK ABWR includes the provision of containment and ventilation systems that are intended to ensure that radioactive substances are retained within designated facilities during normal and fault conditions and that they only enter the environment via appropriately permitted routes. Containment systems are also provided to ensure that radioactive substances do not spread unnecessarily around the plant and generate additional quantities of radioactive waste by contaminating structures, equipment and workers. Collectively, these systems ensure that the radioactivity in discharges from the UK ABWR will be minimised

Containment systems have common objectives related to worker safety and environmental protection which are delivered by effective design, manufacture, installation and operation. Hitachi-GE have developed a series of design policies and principles that aim to ensure that both safety and environmental protection are an inherent part of the design. The Environment Agency REPs [Ref-6] have been considered in the design of containment systems.

The design of the UK ABWR has evolved to enhance the leak tightness of the reactor coolant circuit by rationalising the amount of pipework associated with plant operations and by improving the performance of welds, seals and connections (5.1.10.1. Evidence: Application of Design Features for Leak Tightness). Where contained materials have the potential to be released, systems are provided for detection and containment in segregated ventilation systems and drains that are specific to the characteristics of the waste (5.1.10.2, 5.1.10.3, 5.1.10.4, 5.1.10.5, 5.1.10.6). The design also includes a system for leak detection and isolation (5.1.10.8 Evidence: Leak Detection and Isolation System). Design improvements have led to improvements in leak tightness (5.1.10.9 Evidence: Improvements to Leak Tightness).

Measures adopted to prevent leaks are currently being reviewed as part of the hazard analysis assessment. The UK ABWR has also been designed to minimise the leakage of atmospheric argon into the system which otherwise would have the potential to become activated (5.1.10.7 Evidence: Design Policies to Prevent Atmospheric Argon Leaking into the Coolant System and 5.1.10.10 Evidence: Improvements in Turbine gland seal design). The design features and principles will be further refined as this work is developed. At this stage of GDA the development of design features and principles to be considered during the design development process is considered to represent BAT.

5.1.10.1.Evidence: Application of Design Features for Leak Tightness

The following design features included in the UK ABWR design have been developed by Hitachi-GE to reduce leakage and releases of radioactive material [Ref-14]:

1. extremely stringent leak rate requirements specified for all equipment, piping and instruments and will be confirmed using as-installed helium leak tests of the entire process system;
2. use of welded joints wherever practicable;
3. specification of valve types with extremely low leak rate characteristics (i.e., bellow seal, double stem seal, or equivalent);
4. routing of drains through steam traps to the main condenser; and
5. specification of stringent seat-leak characteristics for valves and lines discharging to the environment via other systems.

Confirmation that these design features can be found in the penetration, piping and joint design specifications is provided in [Ref-90][Ref-91][Ref-92].

5.1.10.2.Evidence: Design Policies and Principles for Leak Tightness in the Liquid Waste System

The items summarised below have been taken into consideration when designing the system for treating liquid wastes and the facilities related to them in order to prevent leakage of liquid radioactive substances from these facilities and to prevent their uncontrolled discharge to the environment [Ref-15]:

- To prevent the occurrence of leaks, the liquid waste system shall be made of suitable materials and shall be provided with tank water level detectors.
- As a general rule, the drain pipes and vent pipes which empty outside of the system shall be provided with lockable valves or the equivalent for closing. However, those which are used with a high frequency shall be provided with a drain leading to tanks, sump pits or the equivalent.
- Should radioactive liquids leak out, there shall be provisions making it possible to detect the leaks promptly and to remove and decontaminate the leaked liquids easily.
- The components of the Liquid Waste Management System (LWMS) shall be provided with secondary containment, such as bunds to prevent spreading of any leaks inside the NPP. Bunds shall be provided on inlets and outlets connected to points outside the NPP to prevent leakage outside of the NPP. Outdoor devices and outdoor pipes shall be designed so that any leaked liquids will be collected inside facilities such as shielding walls or pipe ducts.
- Alarms provided for the tank water level or for leak detection shall be designed to alert either the MCR or the radwaste system control room, so that it will be possible to inform the operators reliably of abnormalities.
- The facilities shall be designed to ensure that facility drainage systems are segregated and do not connect with drainage channels discharging effluents outside of the site in an uncontrolled manner.
- Welded joints are adopted for connections between equipment and pipe work for radioactive waste unless disconnections are required for maintenance.
- Waterproof/impermeable coating is applied to the floor and wall where radioactive waste may leak.

The application of some of these design principles is described in Table 5.1.10.2-1. Further details can be found in the PCSR Chapter 18 [Ref-58].

Table 5.1.10.2-1: Materials of the Radioactive Liquid Waste Disposal and Treatment Facility

Property of retained water	Materials	Typical example
High corrosive potential due to the ion concentration of chloride, sodium, and sulphate being high.	Selection of stainless steel or carbon steel & lining.	Concentrated waste storage tank and pipe work.
Corrosive environment as a result of acid or alkali.	Selection of stainless steel.	HCW collection tank and pipe work.
Low corrosive potential combined with a high crud density.	Selection of stainless steel.	LCW collection tank, sludge storage tank and pipe work.
Low corrosive potential combined with requirement to maintain water quality.	Selection of stainless steel.	Sample tank and pipe work.
Others.	Carbon steel.	Other pipe work.

5.1.10.3. Evidence: Design Policies and Principles for Leak Tightness in the Off-Gas System

Leakage of off-gas into other parts of the NPP is prevented by operating the off-gas charcoal adsorber at negative pressure. As a general rule, the valves in contact with radioactive gases are bellow seal types [Ref-16]. Further details can be found in PCSR Chapter 18. [Ref-58].

5.1.10.4. Evidence: Design Policies and Principles for Leak Tightness in the Containment Vessel

The inner surface of the UK ABWR containment design is lined with a steel plate which acts as a leak tight membrane. All normally wetted surfaces of the liner in the S/P are made of stainless steel. Penetrations through the liner for the drywell head, equipment hatches, personnel locks, piping and electrical and instrumentation lines are provided with seals and leak tight connections. The design basis allowable leakage is 0.4 percent per day from the Primary Containment Vessel (PCV) [Ref-9] (5.1.10.9 Evidence: Improvements to Leak Tightness).

5.1.10.5. Evidence: Design Policies and Principles for Leak Tightness in HVAC System

The air flow inside buildings will be such that the air flows from areas of low radioactive contamination to areas of higher radioactive contamination, and the reverse flow will not be allowed. Contaminated air generated locally will be exhausted through a hood where practicable. To prevent the spread of contaminated air inside a room, the contaminated area will be at a local negative pressure.

For penetrations of piping or trays between rooms where the radioactive contamination classification differs, an air-tight sealing will be provided to reduce air leakage from the room with higher radioactive contamination to the room with lower radioactive contamination. No openings except hatches or doors will be provided between the rooms and an air-tight seal will be provided for the opening provided between the rooms [Ref-17].

5.1.10.6. Evidence: Design Policies and Principles for Leak Tightness in Fuel Pool

In order to prevent leakage of water from the fuel pool, the fuel pool has been designed without any exhaust ports, and an emergency make-up water system will be provided. Water leakage detectors and water-level alarm devices will be provided in order to monitor any possible leakage of the fuel pool water.

The fuel pool and the cask pit do not have drain outlets to prevent the leakage of fuel pool water. The FPC is designed to recirculate water that is released into the skimmer surge tank beyond the skimmer weir. In addition, check valves are placed on the pipes that lead to the fuel pool to avoid the release of fuel pool water by a siphon action [Ref-9].

5.1.10.7.Evidence: Design Policies to Prevent Atmospheric Argon Leaking into the Coolant System

Argon-40, a constituent of air, can leak into the main condenser where it is entrained within the reactor coolant water and passed through the reactor core where it becomes activated forming argon-41. This is then transferred to the steam phase and, being non condensable, is ultimately transferred to the OG via the SJAE where it is then discharged to the environment contributing to radioactive discharges. Measures are taken during the design and manufacture of the main condenser in order to minimise the amount of leakage that occurs, therefore minimising the amount of argon-41 that can be produced.

5.1.10.8.Evidence: Leak Detection and Isolation System

The leak detection and isolation system consists of temperature, pressure and/or flow sensors with associated instrumentation, alarm, and/or isolation functions [Ref-18]. This system detects and indicates and/or alarms following leakage and provides signals to close containment isolation valves, as required, in the following systems:

- Main steam lines;
- Reactor Water Clean-up System;
- Residual Heat Removal System;
- Reactor Core Isolation Cooling System;
- Feedwater System;
- Emergency Core Cooling Systems; and
- Other miscellaneous systems.

An additional radiation sensor with an alarm function is provided for detecting a main steam or pre-filtered coolant leakage. Small leaks are generally detected by monitoring the air cooler condensate flow, radiation levels, equipment space temperature, and drain sump fill-up and pump-out rates. Large leaks are also detected by changes in reactor water level, drywell pressure, and changes in flow rates in process lines.

Manual isolation control switches are provided to permit the operator to manually initiate isolation of the main steam line from the control room. In addition, each Main Steam Isolation Valve (MSIV) is provided with a separate manual control switch in the control room which is independent of the automatic and manual leak detection isolation logic.

The leak detection system provided to detect leakage of coolant from the reactor coolant pressure boundary during normal operation ensures that a leakage of about 3.8l/min [Ref-101] can be detected within one hour. This is achieved by monitoring the volume of condensed water in the gas coolers located in the containment vessel and the water level in the sumps also located in the containment vessel.

5.1.10.9.Evidence: Improvements to Leak Tightness

A number of design improvements have been incorporated into the design of the UK ABWR to prevent leaks which contribute to the generation of radioactive waste [Ref-19].

The MSIVs are designed to isolate the steam line primary containment preventing the loss of coolant and any release of radioactive materials. Previous BWR designs had three MSIVs on the main steam line. The third MSIV controlled and prevented leakage from the first two MSIVs in the case of an accident. However improvements to the valves of the first two MSIVs resulted in a low leak rate. The third MSIV was therefore no longer required along with all the leakage control system associated with it. Figure 5.1.10.9-1 highlights the third MSIV and associated leakage control system that formed part of previous BWR designs that have been removed from the ABWR design as a result of the valve improvements.

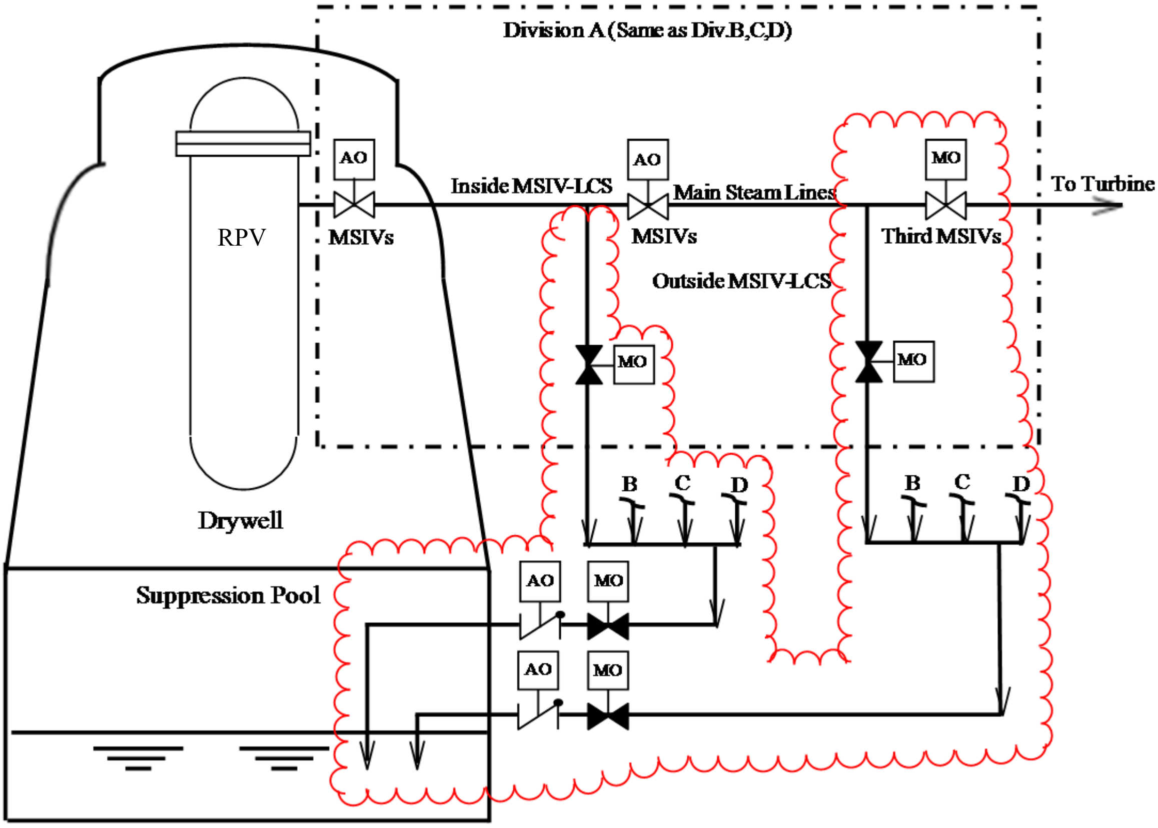


Figure 5.1.10.9-1: Eliminated Equipment from the MSIV Leakage Control System Following Valve Improvements

The improvements to the design and orientation of the first two, MSIVs is detailed in Table 5.1.10.8-1 which led to improved performance of the MSIV negating the requirement for the third MSIV.

Table 5.1.10.9-1: Improvements to MSIVs

Phase	Items	Description
Design	Improvement of disc form	Tighter contact between the two sides of the valves discs.
	Improvement of instalment angle	2 MSIVs were initially installed inclining at an angle of 15 degrees due to space limitations, but installing upright improved seating on the valve.
	Flexible structure of disc	The disc was able to contact firmly to the seat by slitting along the outer periphery disc.
	Improvement of disc lapping process	An automatic disc lapping process rather than a manual process.
Maintenance	Stellite weld overlay at valve seat	The seal properties were improved by Stellite weld overlay at seat side.
	Elimination of difference in level on the seat side	The seal properties were improved by eliminating the difference generated by weld overlay at seat side.

Operational experience and feedback from the existing fleet of BWRs in Japan has demonstrated that leakage rates have reduced to below 5 percent/day. The elimination of the third MSIV and the MSIV leakage control system reduced the worker dose, the costs for maintenance and the radioactive waste during the decommissioning phase.

5.1.10.10.Evidence: Improvements in Turbine gland seal design

A design change was introduced in 1976 to improve the safety and environmental performance of the turbine gland seal. Figure 5.1.10.9-2 shows the comparison of the existing TGS design with the original BWR design. In the original BWR design, the steam seal regulator (SSR) used reactor steam as the gland sealing steam. The reactor sealing steam is collected to the gland steam condenser (GSC), and any steam that is not condensed is then released to the environment via the main stack. This resulted in reactor steam containing elevated levels of radioactive contamination, including noble gases, being released directly to air. The TGS design was replaced by the Separate Steam Seal System (SSSS) and multi-phase gland seal as shown in the same figure which resulted in the following modifications:

- Introduction of a multistage turbine gland (TG) to prevent the loss of containment of reactor steam into the turbine building even under TGS steam failure, and

- Replacement of the reactor water/steam with CST water as the source for the TG seal steam to reduce the source term in the seal steam, including the elimination of noble gases, in order to reduce the radioactivity of gaseous radioactive waste discharges.

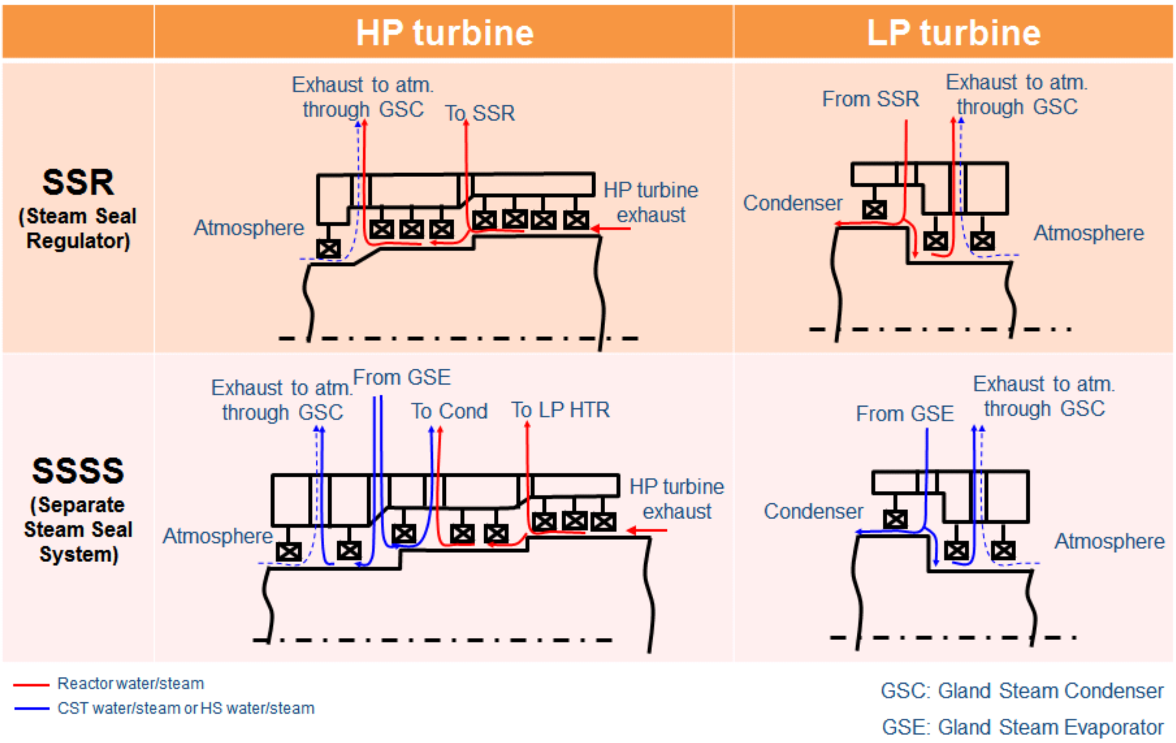


Figure 5.1.10.9-2: Evolution of TGS design

5.2. Claim 2 - Minimise the Radioactivity in Radioactive Waste Disposed to the Environment

The UK ABWR employs a range of features to reduce the discharge or disposal of radioactivity from those radioactive wastes that are unavoidably created during operations.

The Arguments presented in support of this Claim are considered to demonstrate compliance with the standard BAT conditions [Ref-2]:

- Condition 2.3.2(a) 'The operator shall use the best available techniques in respect of the disposal of radioactive waste pursuant to the permit to minimise the activity of gaseous and aqueous radioactive waste disposed of by discharge to the environment.'
- Condition 2.3.3(a) 'The operator shall use the best available techniques to exclude all entrained solids, gases and non-aqueous liquids from radioactive aqueous waste prior to discharge to the environment.'

This is also considered to fulfil the following requirements of the P&ID [Ref-1]:

- Minimising (in terms of radioactivity) discharges of gaseous and aqueous radioactive wastes.

The UK ABWR design contains a range of features that contribute to the substantiation of this Claim including:

- Provision of an OG which includes processes to reduce radioactivity in the gaseous phase prior to discharge to the environment.
- Provision of off-gas charcoal adsorber within the OG to abate short lived noble gasses.
- A Heating Ventilating and Air Conditioning (HVAC) system that prevents the uncontrolled discharge of radioactive substances.
- Treatment techniques for aqueous wastes that minimise the discharge of radioactivity to the environment.
- Delay beds to minimise the radioactivity associated with wastes that require disposal.

In developing the Arguments presented to demonstrate the validity of Claim 2, the following REPs are considered to be relevant and have been taken into account:

- **Principle ENDP14** 'Best available techniques should be used for the control and measurement of plant parameters and releases to the environment, and for assessing the effects of such releases in the environment.'
- **Principle ENDP15** 'Best available techniques should be used to prevent and/or minimise releases of radioactive substances to the environment, either under routine or accident conditions.'
- **Principle ENDP16** 'Best available techniques should be used in the design of ventilation systems.'

5.2.1. Argument 2a: Off-Gas Waste Treatment System

Gaseous radioactive wastes will be generated during the operation of the reactor. Significant efforts are expended to eliminate these wastes but the disposal of gaseous waste to the environment is required to ensure the safe and efficient operation of the NPP. Some of the radionuclides that are carried in the steam do not condense in the condenser. These radionuclides are carried by the Steam Jet Air Ejector (SJAE), which is used to maintain the vacuum in the condenser, and require treatment and disposal as gaseous radioactive waste.

The design of the UK ABWR includes an OG which collect, convey, treat and discharge gaseous radioactive waste from the condenser (5.2.1.1: Evidence: Configuration of the Off-Gas Waste Treatment System). The system includes processes to reduce radioactivity in the gaseous phase prior to discharge to the environment.

A separate Argument is presented for the most significant treatment process which is the decay of noble gases and iodine through the use of delay beds (5.2.2 Argument 2b: Delay Beds for Noble Gases and iodine) and filtration of airborne particulate matter (5.2.4 Argument 2d: Filtration of Airborne Particulate Matter).

Some of the radionuclides in the off-gas such as carbon-14 do not undergo treatment in the OG and are discharged directly to the environment via the stack. Whilst tritium does not undergo dedicated abatement the majority of the tritium in the gaseous phase is removed from the off-gas by the OG recombiner and OG condenser. Dedicated abatement is not provided for tritium and carbon-14 because the assessment of treatment techniques for these radionuclides (5.2.1.3: Evidence: Assessment of Gaseous Treatment Techniques for Tritium and Carbon-14) has shown that the costs of installation and operation are very high and the reduction in impacts on members of the public and the environment is low. Installation of such equipment is therefore considered to be grossly disproportionate to any benefit that would be realised. Hitachi-GE considers that sufficient evidence has been provided to demonstrate that providing abatement for particulate matter and short lived radionuclides and not providing treatment for tritium and carbon-14 represents BAT.

The UK ABWR design provides sufficient space to allow a future operator to carry out in-process monitoring to confirm that what has been identified as BAT is performing as expected.

5.2.1.1. Evidence: Configuration of the Off-Gas Waste Treatment System

The UK ABWR OG reflects RGP from across the nuclear industry and the experience gained from operating previous generations of the BWR and Japanese ABWR. The configuration of the OG reflects this experience ensuring that gaseous radioactive waste discharges into the environment are minimised.

The design of the OG has been developed to address three primary functions:

1. To maintain the main Condenser vacuum by extracting non-condensable gas.
2. The safe recombination of flammable gases (hydrogen and oxygen), which are generated by radiolytic decomposition of reactor cooling water, to reduce the possibility of a hydrogen explosion. Whilst this was not primarily designed to contribute to the environmental performance of the NPS, it does serve as an effective method for removing tritium from the off-gas. This is because tritium is a hydrogen isotope so the hydrogen recombiner also removes tritium from the off-gas. This is achieved as tritiated hydrogen gas is converted to tritiated water which is then condensed by the OG condenser and returned to the Condensate Storage tank where it is reused within the plant [Ref-67]. The specification and performance of this system is addressed in detail within the PCSR Chapter 18. [Ref-58].
3. To minimise and control the release of radioactive gases and particulates into the atmosphere by delaying and filtering the off-gas waste process stream to adequately decay short lived radioactive isotopes and filter out particulate matter.

A number of different techniques are used for the treatment of off-gas from nuclear installations in order to reduce the radioactivity of discharges. The techniques are primarily focussed on holding up short lived radionuclides to allow them to decay and thereby reduce the overall radioactivity discharged to the environment. A summary of the different gaseous short lived radionuclide treatment techniques is provided in Table 5.2.1.1-1.

Table 5.2.1.1-1: Treatment Techniques for Short Lived Radionuclides in the OG

Treatment technique	Description
Compressed gas storage	Following passage through a recombiner the gas is compressed and directed to one of several storage tanks. Compressed gas storage systems achieve a reduction in activity release by storing the FP gases under pressure in large tanks to allow them time to decay. Although systems of this type have been used in previous generations of BWR, compressed gas systems are no longer being proposed for new NPPs because of the large storage volumes required and the operating complexities introduced by the use of a compressor downstream of the SJAE.
Cryogenic distillation	The krypton and xenon are condensed out of the gas stream as it passes through a distillation column operated at very low temperatures. As the process temperature is lowered from ambient conditions, the gases begin to liquefy, with the amount of liquefaction of each gas depending upon its boiling point and vapour pressure. The condensed krypton and xenon are collected in a sump located in the distillation column. When the sump becomes full, the liquefied krypton and xenon are transferred to a gas cylinder for storage and decay prior to eventual release. Distillation systems are capable of very low release rates, but when compared to a refrigerated charcoal system, the gained incremental reduction in release is generally an insignificant part of the total NPP release. The use of cryogenic distillation also adds considerably to construction and operating costs.
Charcoal delay beds	The off-gas is passed through a charcoal bed which will hold the FP gases long enough to allow those having short half-lives to decay. To increase the adsorption efficiency of the charcoal, any water vapour remaining with the gas is extracted by a moisture removal subsystem. The charcoal is contained in several tanks operated in series downstream of the moisture removal equipment.
Cryogenic charcoal	Following the recombiner the gas is cooled and passed through a moisture separator and desiccant dryer which serves to keep ice crystals from plugging the cryogenic charcoal bed. The charcoal bed is maintained at a temperature of approximately -170^{o}C by using the nitrogen gas that boils off a liquid nitrogen bath to remove the heat of adsorption. The -185^{o}C liquid nitrogen bath is used to cool the decontaminated effluent gas from the cryogenic charcoal bed prior to passage through the regenerative heat exchanger which returns it to ambient temperature. The saturated bed is regenerated by passing heated nitrogen gas through the bed and causing krypton and xenon to desorb. The gases are then compressed and stored in gas cylinders prior to release. This technique has similar limitations to cryogenic distillation.

The UK ABWR uses activated charcoal delay beds which are widely used in the nuclear industry and considered BAT by the Organisation for Economic Co-Operation and Development (OECD) report [Ref-6]. (A)BWRs also have years of operating experience of using delay beds. The review of other techniques in Table 5.2.1.1-1 identified significant challenges associated with the other techniques.

Figure 5.2.1.1-1 provides an illustration of the OG for the UK ABWR [Ref-16].

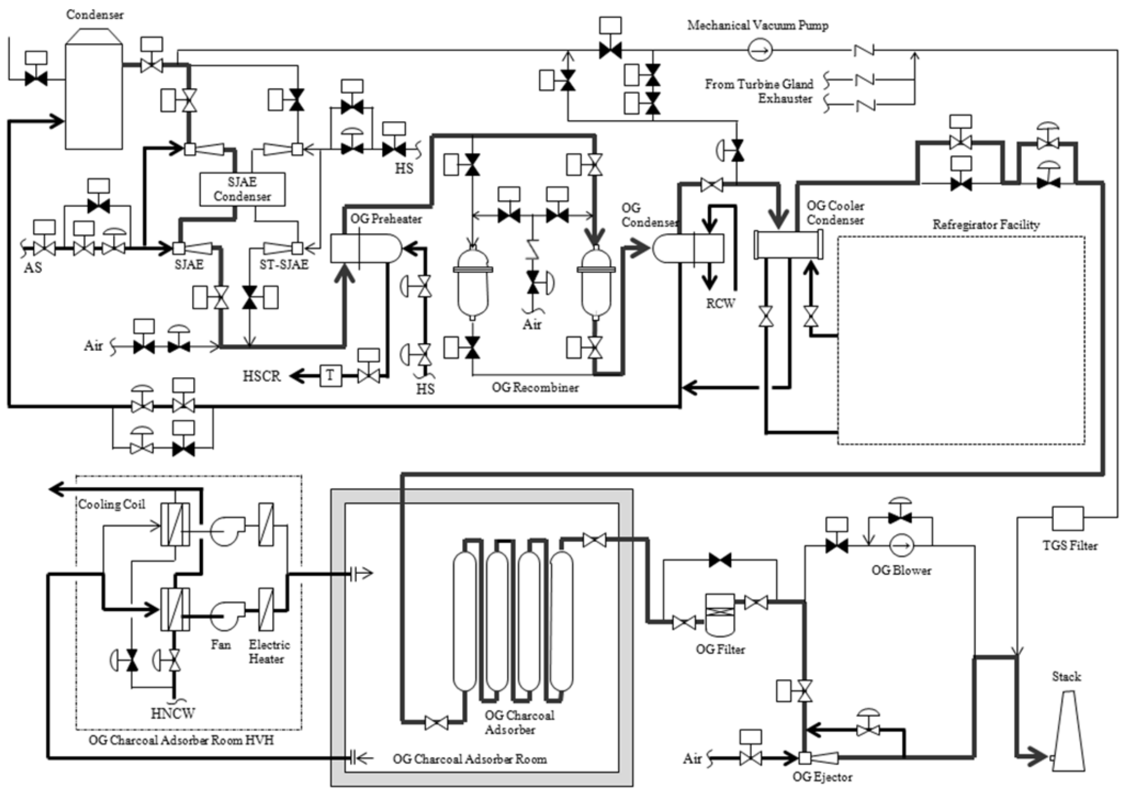


Figure 5.2.1.1-1: Schematic Drawing of the Off-Gas Waste Treatment System

The OG comprises the following unit operations [Ref-16]:

- **1st stage SJAE:** Extracts the off-gas from the main condenser;
- **2nd stage SJAE:** Extracts the off-gas from the main condenser and provides dilution to prevent build-up of hydrogen;
- **SJAE Condensor:** Removes steam between the 1st and 2nd stage SJAE;
- **OG preheater:** Heats up the off-gas to prevent formation of water in the OG Recombiner;
- **OG Recombiner:** Combines hydrogen and oxygen to remove the potential to generate an explosive atmosphere;
- **OG condenser:** Condenses steam to reduce its volume and cool the off-gas;
- **OG cooler condenser:** Cools the off-gas to reduce the moisture content prior to treatment within the OG Charcoal Adsorbers;
- **OG refrigeration facility:** Supplies a cooling source for the OG cooler condenser;
- **OG Charcoal Adsorber:** Holds upshort lived radionuclides (5.2.2 Argument 2b);
- **OG filter:** Removes particulate matter from the off-gas including any particulate from the OG Charcoal Adsorbers (5.2.4 Argument 2d);
- **OG ejector:** Maintains negative pressure within the OG Charcoal Adsorbers;
- **OG blower:** Maintains negative pressure within the OG Charcoal Adsorbers during reactor start-up;
- **OG blower after cooler:** Cools off-gas that has been heated by the OG blower during plant start-up;
- **OG delay bed heating ventilation handling unit:** Controls the temperature of the OG Charcoal Adsorbers room; and
- **TGS filter:** Removes particulate matter from the Turbine Gland Steam (TGS) and Mechanical Vacuum Pump (MVP) discharge.

A more detailed description of the OG is provided in PCSR Chapter 18. [Ref-58].

5.2.1.2. Evidence: Off-Gas Waste Treatment System - Operational Philosophy

The OG is designed to operate continuously in a steady state whilst the UK ABWR is operating. In the event that operational parameters do not remain within specified limits it is normal practice for the operator to shutdown the reactor until the performance of the OG can be demonstrated to have returned to normal operations. The detailed operational philosophy for the OG is described in detail in the OG operational regime document [Ref-51].

5.2.1.3. Evidence: Assessment of Gaseous Treatment Techniques for Tritium and Carbon-14

Tritium and carbon-14 will be present within the gaseous waste that passes through the OG. Delay beds such as those provided in the UK ABWR design are not an appropriate abatement technique for these longer lived radionuclides. An assessment has therefore been carried out by Hitachi-GE to support the application of BAT for the abatement of tritium and carbon-14 [Ref-67].

The discharge rates of tritium and carbon-14 in the gaseous discharge streams during normal operation have been calculated based on the source term [Ref-10], a summary is presented in Table 5.2.1.3-1.

Table 5.2.1.3-1: Summary Table of Total Gaseous Discharges

Radionuclide	**Annual Discharge (Bq/y)**
H-3	**2.7E+12**
C-14	**9.1E+11**

The OECD issued a report in 2003 (Effluent release options from nuclear installations technical background and regulatory aspects [Ref-21]) that stated that "in the particular cases of tritium and carbon-14, there are no abatement techniques in place to reduce discharges". The report identified that carbon-14 and tritium from NPP is discharged without treatment in liquid or gaseous form to the environment and that common practice in many instances is to allow dilution to take place within the plant processes, followed by further dilution and subsequent dispersion upon release to the environment. The report went on to say that tritium removal from gas streams and liquid streams can be uneconomic, especially when concentrations are low.

Hitachi-GE has undertaken an assessment of the treatment options for carbon-14 and tritium [Ref-67]. The techniques assessed are detailed in Table 5.2.1.3-2.

Table 5.2.1.3-2: Options Assessed for the Treatment of Carbon-14 and Tritium

Carbon-14 Treatment Techniques	**Tritium Treatment Techniques**
Alkaline scrubbing	Molecular sieving
Fluorocarbon or ethanolamine absorption	Packed bed or plate columns
Molecular sieve adsorption	
Cryogenic distillation	

The assessment of techniques for the removal of carbon-14 and tritium from gaseous wastes indicate that a number exist, although none are used on operational reactors and are thus not considered to be RGP. In addition to this, International Atomic Energy Agency (IAEA) Technical report No.421 [Ref-22] concluded that methods for the separation of carbon-14 and tritium 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.

The Hitachi-GE study reached the following conclusions;

Tritium

The assessment of tritium abatement techniques identified that none of the treatment processes are currently used on operational LWRs and the techniques would require significant development work to enable the techniques to be suitable for use on an operational reactor. Further to this, very little tritium is actually discharged in gaseous form from the OG as the OG recombiner, which recombines hydrogen and oxygen and the OG condenser, which cools and condenses the hydrogen depleted off-gas to separate any moisture and return it to the main condenser, transfers the majority of the tritium into the liquid phase. As tritium is a hydrogen compound; the performance of the recombiner and OG condenser therefore also removes the tritium from the off-gas. The hydrogen and therefore any tritium is converted to water and is returned to the CST where it is reused within the plant. Recirculating tritium within the process water for the lifetime of the NPP delivers additional benefits from decay.

Carbon-14

The assessment of techniques that have the potential to abate carbon-14 gaseous discharges also identified that none of the treatment processes are currently used on operational LWRs. Based on the evidence available, alkaline scrubbing was identified as a potentially viable technique although a programme of research and development would be required to explore the costs and benefits in sufficient detail to determine its actual viability. At GDA it is considered to be grossly disproportionate to undertake this research and development work on the basis that:

- Although carbon-14 makes a significant contribution to offsite dose, the offsite dose remains very low.
- Development work on alkaline scrubbing treatment techniques has identified significant costs including those associated with the disposal of secondary wastes.

Hitachi-GE has therefore concluded that not abating carbon-14 in the off-gas is BAT at GDA (FA3).

Other measures can be used to minimise tritium production at source rather than implementing treatment techniques [Ref-67] (as summarised in Table 6-1: Summary table of Significant Radionuclides for Gaseous releases). The impacts of the disposal into the environment of both carbon-14 and tritium are minimised by the use of a discharge stack to discharge effluents at height (5.5.1 Argument 5a: Gaseous Discharge System – Main Stack). This ensures that the gaseous effluents emitted are dispersed into the environment in a manner which minimises impacts on members of the public and the environment.

5.2.1.4. Evidence: In-process Monitoring to Support Demonstrating the Application of BAT

Monitoring is carried out in order to ensure that the OG is performing as expected. The performance of the delay beds are influenced by temperature. As a result, temperature in the delay bed room is continuously measured to ensure it is within set parameters. Manual sampling and analysis is also carried out at the delay bed outlet to ensure the radioactivity levels are maintained at a normal level.

5.2.2. Argument 2b: Delay Beds for Noble Gases and Iodine

Low concentrations of FPs such as noble gases and iodine will be present in the off-gas from the reactor cooling circuit due to:

- tramp uranium;
- fission of fuel material;
- fission of structural uranium; and
- ternary fission in fuel;

The concentration of these FPs from within the fuel will increase in the unlikely event of a failure in the fuel cladding. The majority of these FPs have relatively short half-lives and undergo rapid decay. Retention of the FPs in the gaseous waste treatment system for a period prior to discharge reduces the amount of radioactivity that will enter the environment.

The design of the UK ABWRs gaseous waste treatment system includes a series of four delay beds that are filled with charcoal (5.2.2.1: Evidence: Configuration of Delay Beds). The use of such delay beds is common practice in the UK nuclear industry (5.2.2.2 Evidence: Use of Delay Beds). The purpose of these delay beds is to retain the FPs for a defined period during which they undergo radioactive decay. The rate at which the FPs are adsorbed on to and de-adsorbed from the surface of the charcoal is dictated by the chemical and physical properties of the charcoal, the FPs and other species passing through the delay beds. The delay bed system has been designed to retain isotopes of xenon for a period of approximately 30 days and isotopes of krypton for approximately 40 hours. Calculations have been undertaken (5.2.2.3: Evidence: Calculations to Support Delay Bed Size for Xenon and Krypton) that show that the use of the delay beds contributes to a reduction in the amount of radioactive krypton and xenon gas discharged to the environment to 1/30,000. The calculations used to design the delay beds have not specifically taken account of the presence of isotopes of iodine. However, assessment has demonstrated that, as a result of a combination of the high boiling point of iodine and the properties of the charcoal in the delay beds, the delay beds are effective at reducing the amount of iodine that is discharged to the environment (5.2.2.4: Evidence: Calculations to Support Abatement of Iodine).

Evolution of the (A)BWR design has introduced a number of improvements (5.2.2.5: Evidence: Delay Beds for Krypton, Xenon and Iodine - Design Improvements) to the system for retaining noble gases and iodine's. These improvements have increased the length of time that krypton and xenon gases are retained within the gaseous waste treatment system from one day for all gaseous wastes to the current 30 days for isotopes of xenon and 40 hours for isotopes of krypton whilst improving the reliability of the system. They have also contributed to reducing the amount of iodine discharged. Radioactive argon is also present within the off-gas and is held up by the delay beds for 7 hours which achieves a reduction in the amount of argon gas discharged to the environment to 1/14 (5.2.2.6 Evidence: Delay Beds for Radioactive Decay of Argon). The UK ABWR design will include a dedicated temperature controlled room for the delay beds which allows the performance of the system to be maintained whilst reducing the quantity of solid waste generated from drying the off-gases. Further improvements to this system are not considered proportionate on the basis of the DF calculated for the existing abatement system and that any increase in the capacity of the delay beds would require the installation of additional beds. Hitachi-GE considers that sufficient evidence has been provided to demonstrate that the use of delay beds with the stated performance represents BAT. However, to further substantiate this Argument commissioning data shall be used to demonstrate the performance of the delay beds (FA4).

5.2.2.1. Evidence: Configuration of Delay Beds

The OG delay beds comprise four vertical towers (delay beds) filled with charcoal. Each delay bed contains 18 tonnes of charcoal and are connected in series. The OG delay beds receive a flow rate of off-gas of 40m^3/h during normal operations and 80m^3/h during start-up. The flow rate is expected to normally be below the design flow rate of 40m^3/h and, as a result, the off-gas benefits from more hold-up than the design basis calculations shown below [Ref-80]. The number of delay beds and the quantity of charcoal within them is determined by the required hold-up time and the quantity of gas they receive. This is consistent with the Japanese ABWR design.

Evidence supporting the selected hold-up time is provided in Section 5.2.2.3 (Evidence: Calculations to Support Delay Bed Size for Xenon and Krypton). The specification of the charcoal used in the OG delay beds is provided in the UK ABWR design. Confirmation that the charcoal will deliver the required performance will be demonstrated during tests carried out prior to active commissioning. In-process monitoring is provided that will allow a future operator to demonstrate that the activity of gaseous radioactive waste discharges have been minimised. The delay beds are designed for 60 years of operation without a requirement to replace the charcoal. Calculations supported with operational experience from 20 operational NPPs demonstrate that the charcoal will not deteriorate over the 60 years of operation proposed for the UK ABWR [Ref-46].

5.2.2.2. Evidence: Use of Delay Beds

The use of delay beds for the abatement of short lived radionuclides is common practice in the nuclear industry and subject to being appropriately optimised is considered to be BAT, as confirmed in the OECD report [Ref-21]. The OECD report identifies that charcoal delay beds are appropriate for the abatement of noble gases in gaseous discharges and states that they achieve an economically beneficial retention of radioactive noble gases.

Delay beds are used extensively in the nuclear industry and are installed at various UK sites including Sizewell B. They are also provided for the treatment of similar wastes within the design of the PWR proposed for installation at Hinkley Point C.

The Environment Agency [Ref-23] guidance states that "for nuclides with short half-lives that decay to stable (or less hazardous) nuclides, storage prior to discharge represents an option for abatement". This process of decay reduces the activity in the gas and therefore minimises the amount of radioactivity discharged to the environment. Delay beds offer a more passive system (fewer moving parts), overall better safety performance and lower operator dose (less pump maintenance) than alternative techniques. Although delay tanks are a viable option, they offer no gaseous treatment benefits over charcoal delay beds, as discussed in Section 5.2.1.1.

Delay tanks with a hold-up time of 24 hours were used in previous generations of BWR. The OG including the delay tanks provided a decrease in gaseous radioactive waste of 1/300 (5.2.2.3 Evidence: Calculations to Support Delay Bed Size for Xenon and Krypton). Delay tanks have subsequently been replaced with delay beds in the UK ABWR design. This change is considered to be a design improvement that will decrease gaseous radioactive waste discharges to approximately 1/30,000 of the radioactivity at the inlet of the Charcoal Adsorber as demonstrated in 5.2.2.3 (Evidence: Calculations to Support Delay Bed Size for Xenon and Krypton).

5.2.2.3. Evidence: Calculations to Support Delay Bed Size for Xenon and Krypton

The size of the delay beds have been derived using design basis calculations. The calculations used are provided as Calculation 5.2.2.3-1, 2 and 3 [Ref-24].

Calculation 5.2.2.3-1 is used to calculate the quantity of radioactivity for each individual radionuclide at a period in time by inputting the initial quantity of radioactivity and the delay constant.

$$A(t) = A_0 \cdot e^{-\lambda t}$$

where,

$A(t)$: Quantity of radioactivity at time t [Bq/s]

A_0 : Initial quantity of radioactivity [Bq/s]

λ : Decay constant [1/s]

Calculation 5.2.2.3-1: Radioactive Decay Calculation

After calculating the quantity of radioactivity, Calculation 5.2.2.3-2 calculates the DF and then the decay rate. The DF is the ratio of initial specific radioactivity to final specific radioactivity.

$$Decay\ rate = \frac{1}{DF} = \frac{A(t)}{A_0}$$

Calculation 5.2.2.3-2: Decay Rate Calculation

Calculation 5.2.2.3-3 allows for the comparison of the relationship between hold-up time and decay performance.

$$Decay\ rate = \frac{Radioactivity\ at\ t_{(Xe30d+Kr40h)}}{Initial\ concentration\ of\ radioactivity\ (Xe+Kr)} = \frac{3.25E+02\ [Bq/s]}{9.68E+06\ [Bq/s]} = \frac{1}{30000}$$

Calculation 5.2.2.3-3: Decay Rate for Xenon after 30 days and Krypton after 40 hours

The Japanese ABWR provides a hold-up period of 30 days for xenon and 40 hours for krypton [Ref-24]. Using Calculation 5.2.2.3-1, 2 and 3, an assessment has been undertaken to determine if this hold-up period represents BAT for the UK ABWR. The outcome of the assessment is presented within Table 5.2.2.3-1. Table 5.2.2.3-1 demonstrates that radioactive xenon and krypton are reduced to approximately 1/30,000 of the radioactivity at the inlet of the charcoal adsorber using the Japanese ABWR OG (30 days for xenon and 40 hours for krypton). Based on the design basis calculations most benefit is derived during the early part of the retention period and significantly reduces prior to attaining the end of the retention period of 30 days for xenon and 40 hours for krypton. Increasing the retention period would require the inclusion of additional delay beds which is considered to be grossly disproportionate based on the design basis calculations presented in Figure 5.2.2.3-1.

Table 5.2.2.3-1: Radioactive Decay of Xenon and Krypton

Hold-up time for xenon(day)/krypton(hour)	0 (At the Charcoal Adsorber inlet)	1/24	20/26.6	30/40	40/53.3	50/66.7
Quantity of radioactive xenon (Bq/s)	6.81E+06	3.27E+04	9.13E+02	2.45E+02	6.66E+01	1.84E+01
Quantity of radioactive krypton (Bq/s)	2.87E+06	1.16E+03	7.27E+02	7.99E+01	9.90E+00	1.48E+00
Quantity of radioactive xenon + krypton (Bq/s)	9.68E+06	3.39E+04	1.64E+03	3.25E+02	7.65E+01	1.99E+01
DF for xenon + krypton	-	2.86E+02	5.90E+03	2.98E+04	1.27E+05	4.86E+05

The quantity of charcoal in the OG charcoal adsorber that is required to hold up xenon and krypton contained in the off-gas for the established periods (xenon: 30 days, krypton: 40 hours) is calculated by Calculation 5.2.2.3-4 and Calculation 5.2.2.3-5.

$$M = \frac{W \times tH}{K} \times (1 + a)$$

Where:
M: Quantity of charcoal (t)
W: off-gas flow rate (m^3/h)
tH: Hold-up time (h)
K: Dynamic adsorption coefficient (m^3/t)
a: Margin ratio (-)

Calculation 5.2.2.3-4: Required Charcoal Quantity Calculation

$$Kr = M = \frac{56.4 \times 40}{36} \times 1.15 = 72.02 = 72 \text{ (t)}$$

$$Xe = M = \frac{56.4 \times 30 \times 24}{650} \times 1.15 = 71.84 = 72 \text{ (t)}$$

Where:

OG flow rate: 56.4 m^3/h calculated using the normal flow rate ($40m^3/h$) but with the actual temperature (25°C) and pressure (78.45kPa)

Calculation 5.2.2.3-5: Required Charcoal Quantity Calculation

The results of calculations 5.2.2.3-4 & 5 demonstrate that 72 tonnes of charcoal adsorber is required to achieve the required hold-up time of 30 days for xenon and 40 hours for krypton.

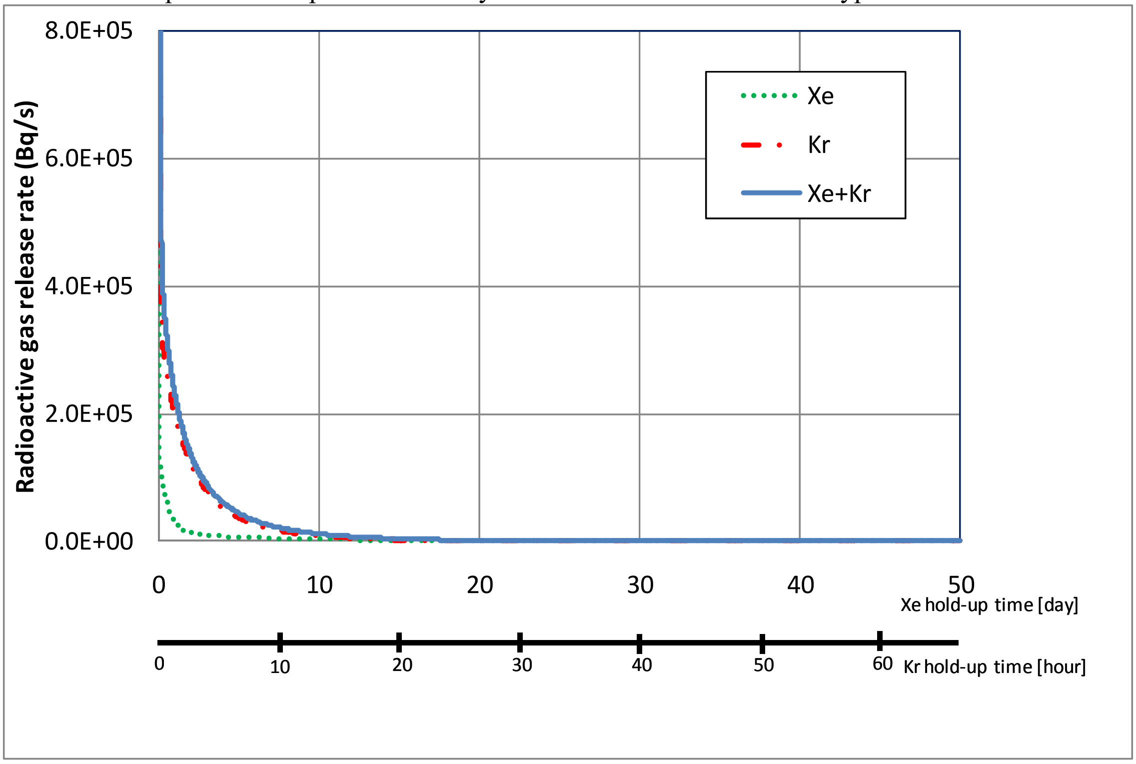


Figure 5.2.2.3-1: Hold-up Time versus Release Rate for Krypton and Xenon

5.2.2.4. Evidence: Calculations to Support Abatement of Iodine

Radioactive iodine is mostly retained within the condensate of the main condenser as a result of its high boiling point. As such, little radioactive iodine is carried over into the OG. Whilst the OG delay beds are designed to hold-up radioactive xenon and krypton it has been calculated that it also facilitates the decay of any radioactive iodine that is carried over from the main condenser. This is because radioactive iodine has favourable characteristics that promote its retention within the OG delay beds.

The capacity of the OG delay beds to adsorb gaseous substances is effected by dynamic adsorption equilibrium constants of the charcoal. Through experimentation the charcoal adsorption capacity for iodine and methyl iodine has been determined and the specifications of the delay beds are presented in

Table 5.2.2.4-1 [Ref-25].

A separate study was conducted to review abatement techniques used in the nuclear industry for iodine and to support the demonstration that the charcoal delay beds provided in the UK ABWR design are BAT. The review drew upon conclusions from the OECD report [Ref-21] that "volatile iodine FPs from nuclear reactors are abated using carbon filter beds. The filter beds are porous which provide a very high effective surface area for adsorbing the iodine gas. It compared this to a number of techniques such as:

- Silver based solid sorbents;
- Cadmium based solid sorbents;
- Lead based solid sorbents;
- Alkali liquids;
- Iodox liquid;
- Mercurex liquid;
- Electrolyte scrubbing;
- Fluorocarbon Absorption; and
- Organic solvents.

It concluded that, although there are other techniques available which can abate iodine, no other techniques have been used on NPPs other than carbon delay beds. Other techniques are used on fuel reprocessing facilities on the basis that carbon is not feasible for use. This is due to issues with other techniques such as:

- Costs;
- Inferior abatement when compared with carbon delay beds;
- Large volumes of secondary wastes produced; and
- Additional safety concerns associated with handling corrosive and/or chemically toxic agents.

UK ABWR delay beds are also required to abate a number of short lived radionuclides in addition to iodine and carbon delay beds are effective at abating these radionuclides. Additionally, carbon delay beds are considered technically mature. To this effect, the cost of further enhancement of abatement techniques, or the development of new techniques for the UK ABWR would not be justified or cost effective. As such, it was considered BAT to employ carbon delay beds for the abatement of iodine.

Further to this, the IAEA publication [Ref-45] also states that for NPPs "elemental iodine is generally removed by a physical adsorption process and NNPs almost exclusively use activated carbon for the removal of radioiodine".

Table 5.2.2.4-1: Specification of Standard UK ABWR Delay Beds

Radioactive Gas	**iodine-131**		**iodine-133**	
	iodine	**methyl iodine**	**iodine**	**methyl iodine**
Activated charcoal quantity (M) [t]	72			
Off-gas flow rate (W) [m^3/h]	56.39			
Dynamic adsorption equilibrium constant (K) [m^3/t]	5.16E+04	9.22E+03	5.16E+04	9.22E+03
Hold-up time (h)	6.58E+04	1.18E+04	6.58E+04	1.18E+04
half-life (h)	1.93E+02	1.93E+02	20.80	20.80
Decay constant (λ) [/h]	3.59E-03	3.59E-03	3.33E-02	3.33E-02

Radioactive Gas	iodine-131		iodine-133	
	iodine	**methyl iodine**	**iodine**	**methyl iodine**
DF	5.20E+102	2.36E+18	infinity	2.80E+170

The DF for iodine and methyl iodine is greater than that of xenon and krypton whilst the quantities within the off-gas are lower. This means that the OG charcoal adsorber has a large capacity to decay iodine and methyl iodine. Thus it is reasonable to conclude that discharges of iodine and methyl iodine to the environment from the OG will be very low.

5.2.2.5. Evidence: Delay Beds for Krypton, Xenon and Iodine - Design Improvements

Figure 5.2.2.5-1 provides an illustration of the changes introduced in the 1970's and 1980's to improve the performance of the OG [Ref-13]. This change resulted in the replacement of the delay tanks with charcoal delay beds. The delay tanks provided a minimum hold up period of 1 day and operated as a batch process. The delay beds deliver a minimum hold-up of 30 days for xenon and 40 hrs for krypton and operate continuously.

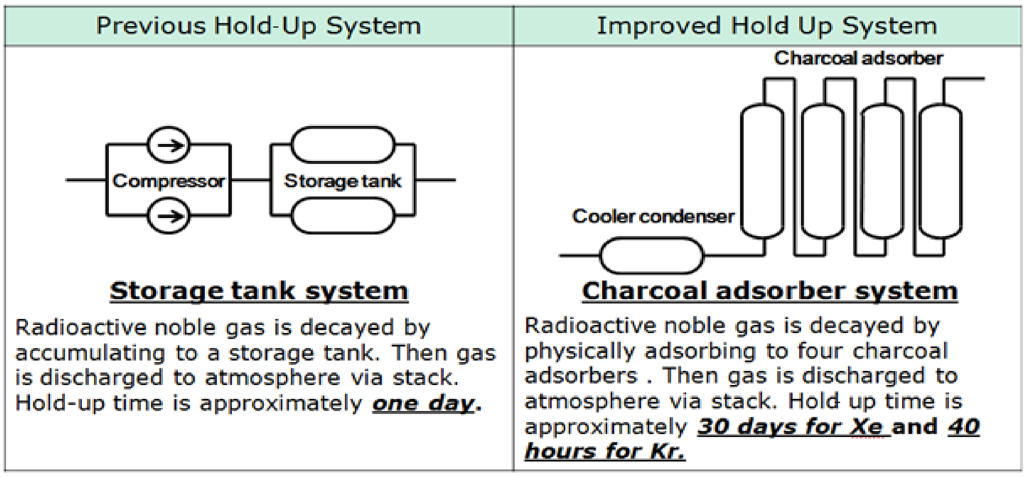


Figure 5.2.2.5-1: Design Improvements to the Off-Gas Delay Bed System

The result of increasing the hold-up time by providing delay beds resulted in a significant decrease in the concentration of radioactive krypton and xenon gases released to the environment. A comparison of discharges from each system is provided in Figure 5.2.2.5-2.

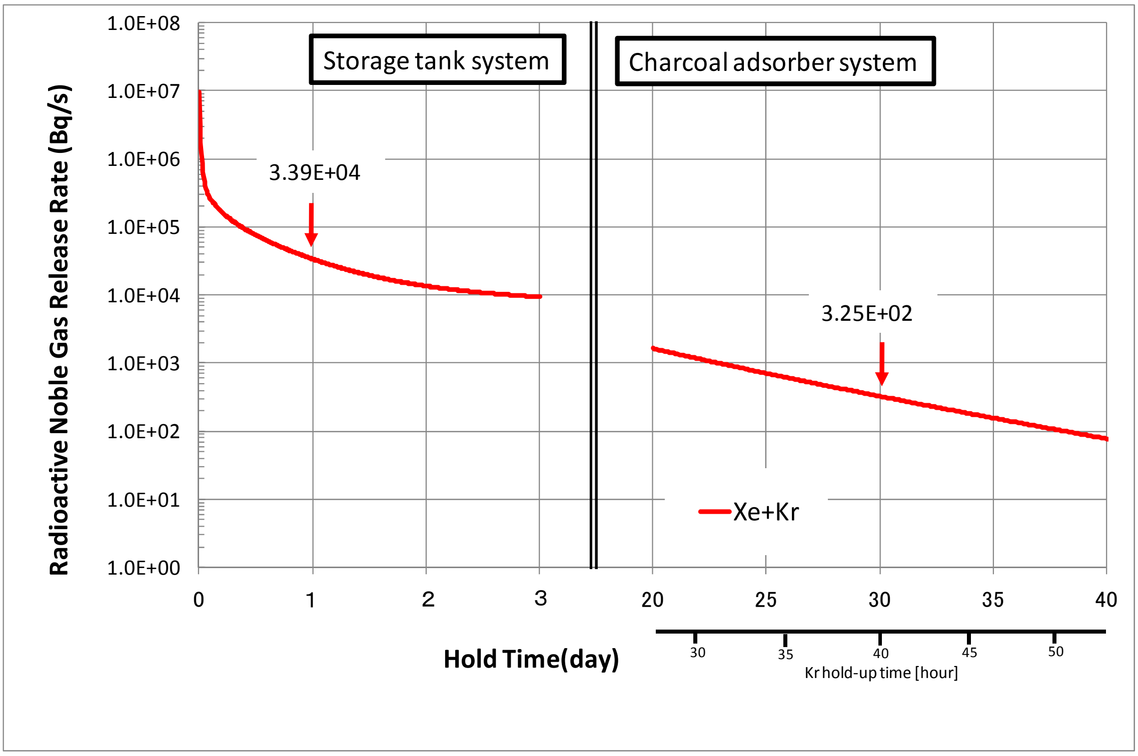


Figure 5.2.2.5-2: Release Rate of Radioactive Krypton and Xenon Over Krypton and Xenon Gas Hold Time Following Design Improvements

5.2.2.6. Evidence: Delay Beds for Radioactive Decay of Argon

As detailed in section 5.2.2.3 the primary focus of delay beds is for the radioactive decay of krypton and xenon. Section 5.2.2.4 describes how there is also even greater decay benefits for radioactive iodine. Argon-41 is also present within the OG as a result of activation of argon-40 which is a constituent of air which leaks into the main condenser. Measures are taken during manufacture of the main condenser in order to minimise the amount of leakage that occurs, therefore minimising the amount of argon-41 that can be produced.

The charcoal delay beds offer 7 hours of hold up time for argon-41 which achieves a reduction in radioactivity of 1/14 of its activity prior to abatement as calculated in Table 5.2.2.6-1 and shown in Figure 5.2.2.6-1.

Table 5.2.2.6-1: Radioactive Decay of Argon

Hold-up time for argon (hour)	**0 (At the Charcoal Adsorber inlet)**	**4.7**	**7**	**9.3**	**11.7**	**24**
Quantity of radioactive argon (Bq/s)	8.14E+05	1.39E+05	5.72E+04	2.36E+04	9.73E+03	9.03E+01
DF for argon	-	5.86E+00	1.42E+01	3.45E+01	8.37E+01	9.01E+03

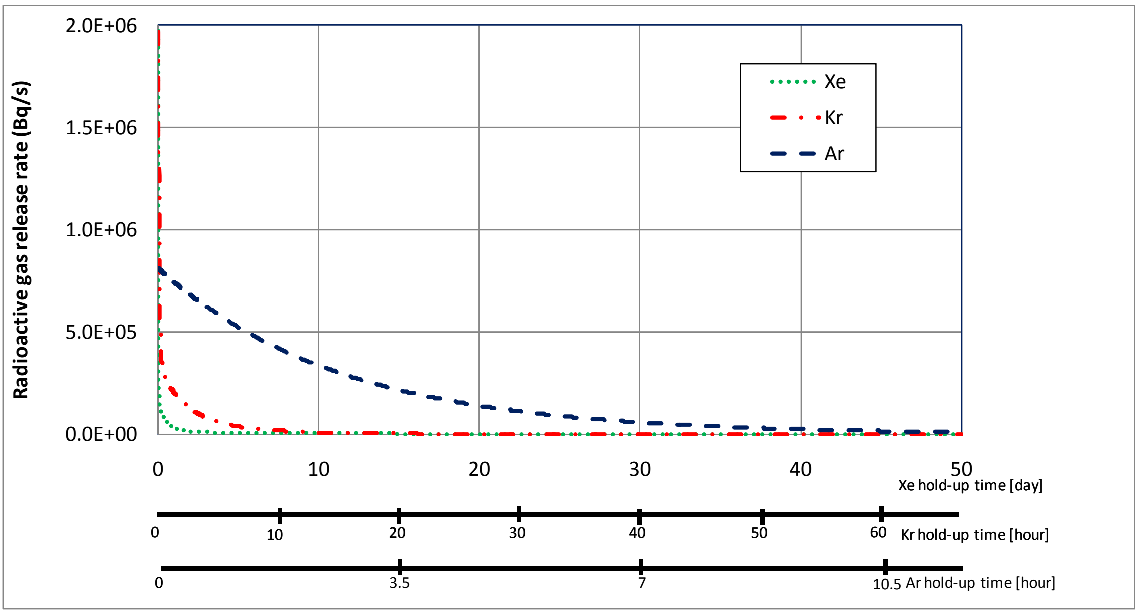


Figure 5.2.2.6-1: Hold-up Time versus Release Rate for Argon, Krypton and Xenon

As displayed in Table 5.2.2.6-1, the DF achieved for argon is much greater following the 24 hours of decay that would have been achieved in the old delay tank design than under the 7 hours of decay in the more recent delay bed design. Whilst the delay bed does not perform as well for argon as the delay tank, the design change represented an improvement in a number of other aspects:

- Hazard reduction – delay tanks are high pressure systems which increase the likelihood of a leak whereas the charcoal adsorber operates under negative pressure so, even in the event of a crack or rupture, the holdup system does not leak.
- Reliability – the delay tank operates in batches with repeated start and stop operations and include a number of valves, whilst the delay beds operate in a steady state mode.
- Performance improvements - The delay beds offer much greater abatement of krypton and xenon using more reliable and passive technology, particularly following a fuel failure where levels of radioactivity increase.
- Reliability – Less risk of operator error due to a passive, continually operating system.

Options to further optimise the abatement of argon-41 have been identified within Table 5.2.2.6-2.

Table 5.2.2.6-2: Options to Further Optimise Ar-41 Abatement

	Advantages	Disadvantages
Zeolite Adsorber	Zeolite has a uniform fine pore diameter which can selectively adsorb particular gases, for example argon.	Installation of an additional zeolite filter would be costly and would require additional space in the T/B. In addition, the zeolite filter would require periodic replacement which would result in an increase in secondary waste arisings.
Introduction of a storage tank in addition to the existing delay beds	A storage tank could provide further decay time for argon-41.	Additional cost of installation and high space requirements in the T/B. Reintroduction of the disadvantages associated with delay tanks.
Increase number/ size of delay beds to increase residence time to 24 hrs	More delay beds would provide an increased delay time and therefore improve the abatement of argon-41.	Additional cost of installation and high space requirements in the T/B resulting in the requirement to provide approximately 247 tonnes ($24/7 \times 72$) of additional charcoal in order to achieve the equivalent of a 24hr hold up period.

It has been demonstrated that the existing delay beds demonstrate an improvement for the abatement of the off-gas beyond that achieved by the delay tanks provided on earlier BWR designs. However, it is recognised that the delay tanks have the potential to delay argon-41 for a longer period than that offered by the delay beds provided in the ABWR design. The transition to delay beds overcame a number of disadvantages associated with the operation and safety of delay tanks and improved the reliability and performance of abatement for short lived radionuclides contained within the off-gas.

The review of options that could potentially increase the abatement of argon-41 identified that a dedicated abatement system would need to be provided in the form of alternative technology such as a zeolite based adsorber, reintroduction of delay tanks or increasing the number of delay beds.

Qualitative assessment of the additional techniques for further abatement of argon concludes that all three options would be prohibitively expensive compared to the potential benefit and would require an extension to the T/B. Owing to the fact that argon-41 only contributes a small amount of the dose [Ref-10], it is considered grossly disproportionate to abate argon further than the abatement achieved in the delay beds.

5.2.3.Argument 2c: Heating, Ventilation and Air Conditioning System

Gaseous radioactive waste will be discharged to the environment via appropriately permitted outlets. The air pressure in facilities handling radioactive substances are typically maintained at a lower level than atmospheric pressure to ensure that air flows into the facility from the external environment. This prevents the uncontrolled discharge of any radioactive substances through doors, windows and gaps in the building fabric. The negative pressure within the facility is maintained by a ventilation system which continuously extracts air from within the NPP facility buildings and discharges it to the environment during normal operations.

The design of the UK ABWR includes a HVAC system which delivers the combined function of:

- preventing the uncontrolled discharge of radioactive substances;
- providing a pleasant working environment for workers;
- ensuring optimal working conditions for plant and equipment; and
- delivering safety related functions to protect workers in the event of a release of radioactivity.

The configuration of the HVAC system ensures independent operation of sub-systems in principle areas of the plant (5.2.3.1: Evidence: Configuration of HVAC System). It also makes efficient use of the air that is drawn in to the system by allowing it to flow from areas of lower contamination risk to areas of higher contamination risk. HVAC sub-systems that serve areas of the plant where radioactive substances are present have filters to remove any particulate matter prior to discharge to the environment via appropriately permitted outlets. The approach to selecting a filter type that is appropriate for the characteristics of the gaseous waste is described in a separate Argument (5.2.4 Argument 2d: Filtration of Airborne Particulate Matter).

HVAC sub-systems that serve areas of the plant where radioactive substances are handled do not provide any abatement other than filters. Additional abatement systems are not considered necessary because, under normal operations, the amount of radioactivity that is expected in the large volumes of waste air drawn through the HVAC system will be very low (5.2.3.2 Evidence: HVAC discharges). Hitachi-GE considers that sufficient evidence has been provided to demonstrate that the HVAC system philosophy represents BAT. To support the demonstration that those parts of the design that have been identified as BAT are operating as expected the UK ABWR design provides sufficient space and access to allow a future operator to carry out in-process monitoring (5.2.3.3 Evidence: In-process Monitoring to Support Demonstrating the Application of BAT in the HVAC System).

5.2.3.1. Evidence: Configuration of HVAC System

The HVAC System will be provided to maintain environmental conditions within the NPP and provide a cascade air flow from areas of low contamination to areas of higher contamination.

The HVAC system will be segregated into sub-systems according to the principle areas and functions of the NPP. The key HVAC sub-systems include those provided for the:

1. Reactor area;
2. Emergency diesel generator electrical equipment area;
3. Turbine area;
4. Heat exchanger area;
5. Instrument and control power supply panel area inside control building;
6. MCR;
7. Backup Building ECR;
8. Backup Building diesel generator electrical equipment area;
9. Controlled area inside Radwaste Building (RW/B); and
10. Service Building (S/B).

The HVAC system has been designed to ensure that appropriate abatement is provided on those systems that serve areas that have the potential to generate gaseous radioactive waste, these are summarised below and further detail is provided in PCSR Chapter 16 [Ref-17]. Ultimately all HVAC systems discharge via appropriately permitted outlets.

1. HVAC System for reactor area

The configuration of the reactor area HVAC system is shown in Figure 5.2.3.1-1. This system provides ventilation and air-conditioning to the reactor area within the Reactor Building (R/B) during normal operations, and this system consists of a supply and exhaust air treatment facility. Air is supplied to the reactor area and the exhaust air collected from each area is discharged from the main stack via the exhaust air treatment facility by the exhaust fans. Filtration will be provided on this system to minimise discharges of radioactive particulate matter. HEPA filters that are compliant with UK standards such as NVF/DG001 (An Aid to the Design of Ventilation of Radioactive Areas) [Ref-26] will be installed in UK ABWR. Two stages of multiple safe change High Efficiency Particulate Air (HEPA) filter units will be provided. The requirement for two stages of filtration was primarily driven by the area classification; during operations the reactor area is a C2 area however this rises to C3 during maintenance.

A small amount of water vapour is expected to be released via the HVAC system as a result of evaporation from the SFP. This water vapour may contain some radioactivity, the majority of which is made up of tritium. The FPC maintains the water quality of the SFP and heat exchangers remove the decay heat and reduce the temperature of the water. The water temperature within the SFP is maintained below 52°C for operability reasons and the safety of operators. Maintaining the water temperature below this limit also reduces the evaporation rate and therefore discharges via the HVAC. Temperature monitoring is provided in the SFP to ensure that temperatures remain below this limit.

2. HVAC System for T/B

The configuration of the T/B HVAC system is shown in Figure 5.2.3.1-2. This system provides ventilation and air-conditioning to the turbine area within the T/B during normal operations, and this system consists of a supply and exhaust air treatment facility. Air is supplied to the T/B, and the exhaust air is discharged from the main stack to the environment. Exhaust air from potentially contaminated areas of the T/B or component vents is collected, filtered and discharged to the atmosphere through the T/B high radiation area exhaust system. HEPA filters that are compliant with UK standards such as NVF/DG001 [Ref-26] are installed in UK ABWR. Two stages of multiple safe change HEPA filter units are provided in the C2 area of the T/B which becomes a C3 area during maintenance and a single stage of HEPA filtration is provided in the area of the T/B which remains C2 during maintenance.

Figures 5.2.3.1-1 to 5.2.3.1-2 show the configuration of the HVAC systems for the reactor area and T/B.

3. HVAC System for Controlled Area inside the RW/B

The HVAC system for the controlled area provides ventilation and air-conditioning within the RW/B during normal operations, and this system consists of a supply and exhaust air treatment facility. Air is supplied to the RW/B, and the exhaust air collected from each area is discharged from the main stack via the exhaust air treatment facility by the exhaust fans. Filtration will be provided on this system to minimise discharges of radioactive particulate matter. HEPA filters that are compliant with UK standards such as NVF/DG001 [Ref-26] are installed in the UK ABWR. Multiple safe change HEPA filter units are provided, and one stand-by housing is installed.

4. HVAC System for S/B

The configuration of the S/B HVAC system provides ventilation and air conditioning within the S/B during normal operations, and this system consists of a supply and exhaust air treatment facility. Air is supplied to the S/B, and the exhaust air collected from each area of the S/B is discharged from local exhaust opening via the exhaust air treatment facility by the exhaust fans. HEPA filters that are compliant with UK standards such as NVF/DG001 [Ref-26] are installed in UK ABWR. Two stages of multiple safe change HEPA filter units are provided in the C2 area of the S/B which becomes a C3 area during maintenance and a single stage of HEPA filtration is provided in the area of the S/B which remains C2 during maintenance.

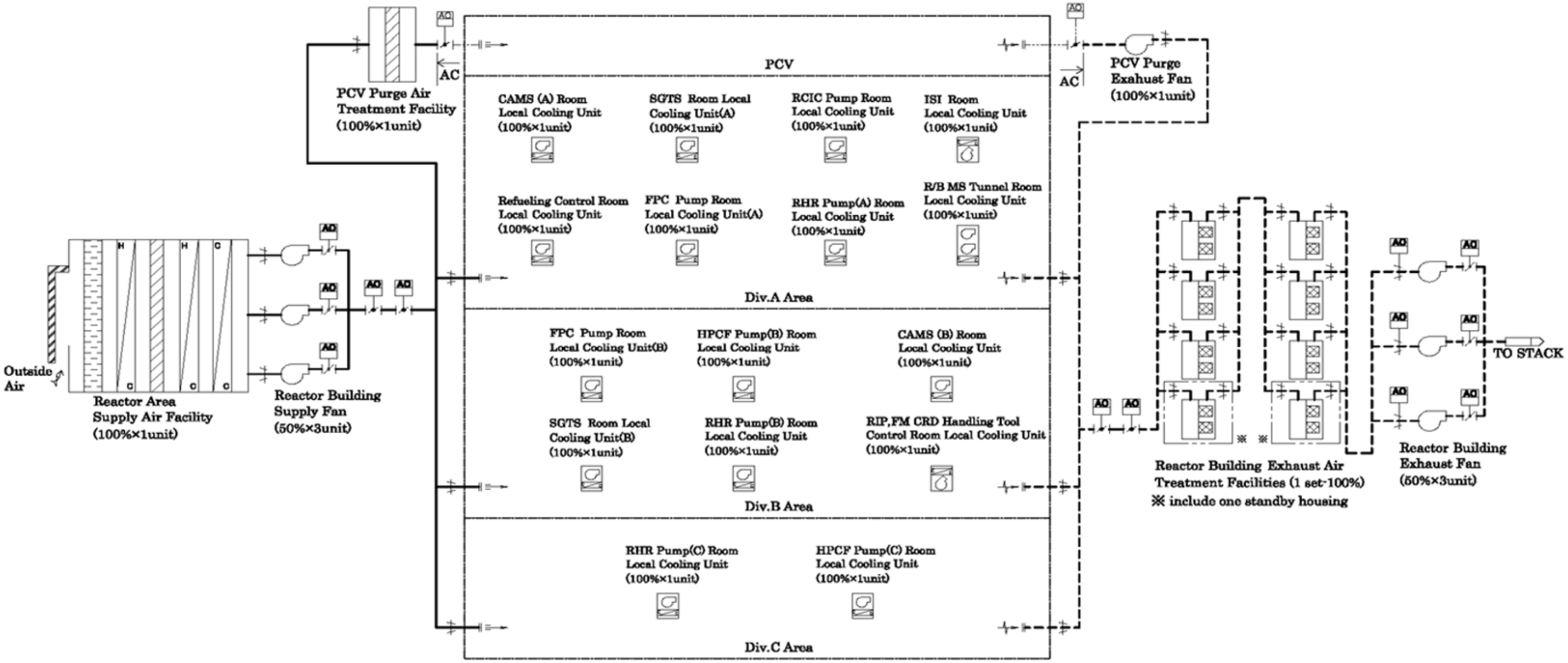


Figure 5.2.3.1-1: Outline of the Reactor Area HVAC

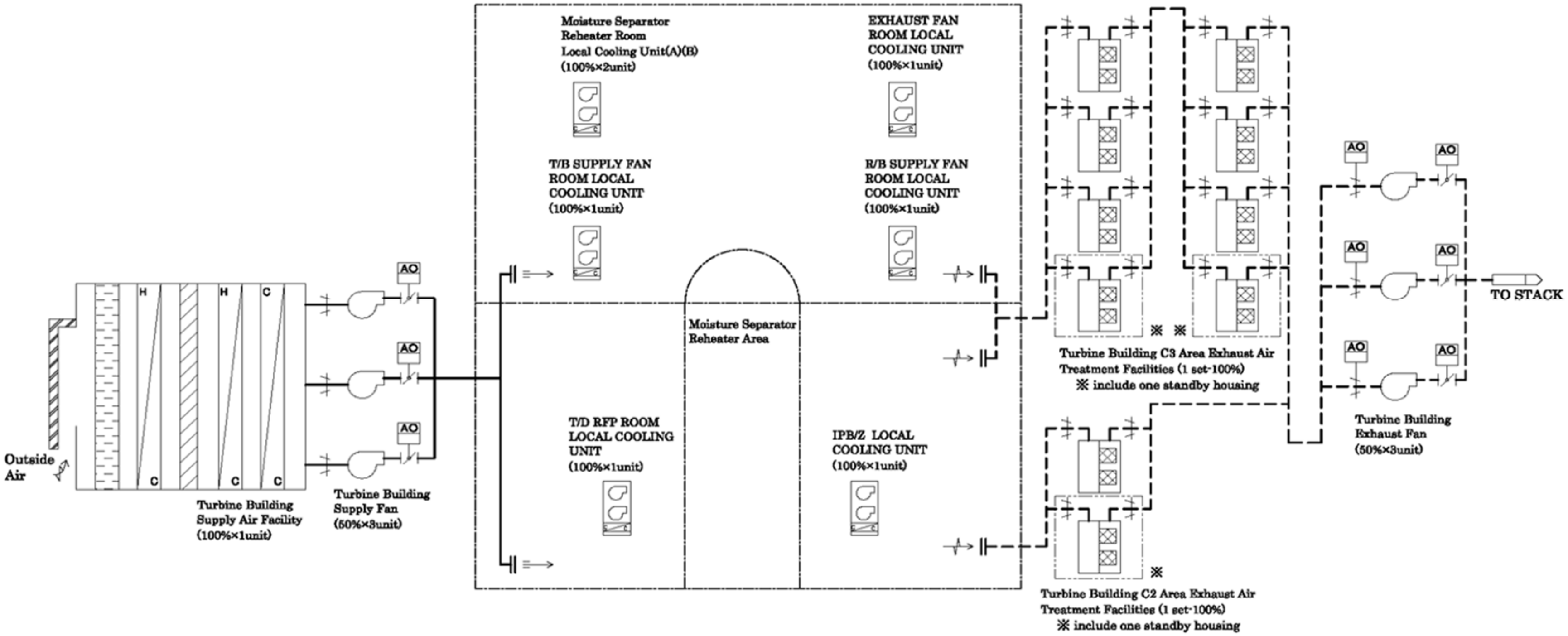


Figure 5.2.3.1-2: Outline of the T/B HVAC

5.2.3.2. Evidence: HVAC Discharges

The realistic radioactive discharge to atmosphere from the HVAC systems serving each building has been estimated by identifying the main radioactive discharge sources (such as SFP) in the source term topic report [Ref-10]. Dose rates from the HVAC system were compared with and without the use of HEPA filters [Ref-60]. Dose rates from HVAC discharges when HEPA filters are not installed are approximately 2.2μSv/y. In comparison, dose rates from HVAC discharges with HEPA filters are approximately 2.1μSv/y. Thus it can be seen that the use of HEPA filters provides a very small benefit in terms of dose reduction during normal operation, however HEPA filters will be installed in the UK ABWR that are compliant with UK standards such as NVF/DG001 [Ref-26].

Hitachi-GE also conducted a study into possible abatement techniques within the HVAC system for iodine. However, it was concluded that the limited source term, high dilution, secondary waste generation and high capital costs for abatement techniques were grossly disproportionate [Ref-81].

5.2.3.3. Evidence: In-process Monitoring to Support Demonstrating the Application of BAT in the HVAC System

Monitoring is carried out in order to ensure that the HEPA filters are performing as expected during normal operations and will perform as expected in accident conditions. In situ differential pressure (DP) monitoring will be carried out by a future operator at the HEPA filter inlet and outlet to check they are performing within defined parameters. DP is used to indicate whether filters are blocking and need changing. Filtration performance is usually measured by in-situ challenge testing (NVF/DG001 [Ref-26] provides details). Stack monitoring after the filters will also demonstrate compliance with limits and identify filter failure (in addition to DP measurement).

5.2.4. Argument 2d: Filtration of Airborne Particulate Matter

The UK ABWR will employ appropriate filtration techniques to ensure that the concentration of particulate matter within the gaseous radioactive waste stream is minimised during normal and accident conditions. Filtration is considered to be RGP in the UK nuclear industry for the abatement of particulate matter and it is the design intent to provide filtration on the UK ABWR HVAC system as appropriate (5.2.4.1: Evidence: Application of Filtration in the Nuclear Industry).

The UK ABWR has been subject to considerable optimisation that has resulted in the amount of particulate matter that has the potential to become mobilised within the building areas served by the HVAC systems being minimised. As a result, during normal operations, concentrations of particulate matter are not expected to be significant. The number and specification of the filters to be used to abate airborne particulate matter has been determined through the application of BAT however filter selection will be influenced by the performance requirements being established to mitigate accident conditions (5.2.4.2: Evidence: The Basis for Filter Selection). It is therefore deemed that the performance of the filters will exceed that required for normal operations. Attempts were made to optimise the air flow rates within the HVAC systems in order to reduce the number of filter changes. However, air flow rates are primarily determined based on reducing risk and controlling contamination (5.2.4.3 Evidence: Filtration of Airborne Particulate Matter – Air Flow Rate).

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5.2.4.1. Evidence: Application of Filtration in the Nuclear Industry

The standard technique for the removal of particulate matter from gaseous effluents collected by HVAC systems in nuclear installations is the use of HEPA filters (filters that have a capture efficiency of 99.99% as stated in Technical Guidance from the Environment Agency and the Nuclear Industry Safety Directors Forum [Ref-23] [Ref-26]). During normal operations, particulate matter containing fission or activation products will be abated by filtration. However, the amount of radioactive particulate requiring abatement during normal operation is relatively small. The use of HEPA filtration for HVAC systems in nuclear installations is considered to be RGP and will form part of the nuclear safety case for the UK ABWR.

HEPA filtration is an example of 'best practice' gaseous filtration techniques implemented within the UK ABWR design. HEPA filters will be provided to abate gaseous radioactive waste generated in the UK ABWR where it is demonstrated to be BAT. The HEPA filter type chosen for each system will depend on the environmental and flow-rate conditions anticipated. The majority of filters used to treat process and space extract gaseous wastes are expected to be Commercial Off-The-Shelf (COTS) systems. HEPA filters are one of several 'best practice' techniques used for gaseous radioactive waste abatement throughout the international nuclear industry.

The HEPA filtration systems for UK ABWR meets the industry codes of practice for testing, design and operation with particular reference to UK standards such as NVF/DG001 [Ref-26]. It can be considered that the systems used for UK ABWR are consistent with those used for existing installations in the UK and employed on most other nuclear installations internationally.

HEPA filters will be changed, where practicable, based on performance as determined by using continuous measurement of differential pressures. This means that the filters are not changed at a predefined frequency which can result in the generation of additional volumes of solid waste. This will ensure that filters will be used to their design capacity.

5.2.4.2. Evidence: The Basis for Filter Selection

HEPA filters will be used on the exhaust air treatment facility of the reactor area, T/B, Rw/B and S/B HVAC systems. They are circular safe change HEPA filters as displayed in Figure 5.2.4.2-1.

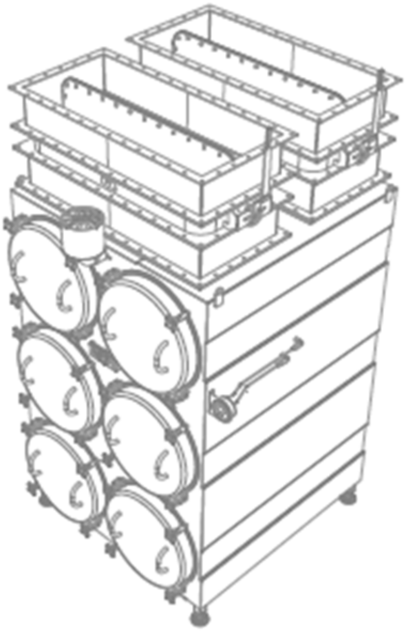


Figure 5.2.4.2-1: Safe Change HEPA filter unit Outline Drawing

Banks of HEPA filters are commonly used to remove radioactive solid particulate matter in dry atmosphere environments. HEPA filters achieve DFs of greater than 1,000 when several filters are arranged in series and parallel combinations. These technologies are relatively mature and it is unlikely that new techniques will significantly enhance their effectiveness [Ref-21].

Exhaust air from the potentially contaminated areas is discharged from appropriately permitted outlets. The extraction system for C2 areas (low contamination areas) includes a single stage HEPA filter whilst the extraction system for C3 areas (high contamination areas) includes two stages of HEPA filters [Ref-17].

5.2.4.3. Evidence: Filtration of Airborne Particulate Matter – Air Flow Rate

Previously, attempts were made to reduce the number of air changes in the UK ABWR HVAC system compared to previous BWR and ABWR designs [Ref-27]. The design modifications that were made to achieve this were:

- the introduction of a refrigerator for cooling supply air;
- reduction of design air change rate; and
- reduction of building capacity.

These design modifications resulted in a reduction in the air flow rate which had the potential to reduce the number of filter changes required and thus would reduce the volumes of solid waste generated. Another benefit would be a reduction in energy consumption as a result of reduced demand on the fans.

Following further review and ALARP assessment, as well as a review of compliance with NVF/DG001[Ref-26], it was concluded that air flow rates should be increased for the UK ABWR. The primary driver for this increase was contamination control achieved by controlling the number of air changes and the velocity requirement for the flow from C2 areas to C3 areas. The flow rates are shown in Table 5.2.4.3-1.

Table 5.2.4.3-1: Reduction in Air Flow Rate from the BWR Design, ABWR Design and UK ABWR Design

NPP Area	BWR generation 5	ABWR (K-6/7)	UK ABWR
Reactor Area	225,000 m^3/h	170,000 m^3/h	245,000 m^3/h
T/B	400,000 m^3/h	328,000 m^3/h	379,000 m^3/h
RW/B	185,000 m^3/h	80,000 m^3/h	157,000 m^3/h

Whilst attempts were made to optimise air changes in order to reduce the generation of solid waste, the air flow rates are driven by the safety function of the HVAC system.

A small benefit of increasing the air flow rate to similar rates found in generation 5 BWRs is an increase in the degree of dilution that will be achieved when the HVAC extract is combined with the treated off-gas prior to being discharged from the stack. As shown by Section 5.5.1.3 Evidence: Gaseous Discharge – Dilution Factor, the dilution achieved is significant and supports the demonstration that the impact associated with discharges to air have been optimised.

5.2.5. Argument 2e: Optimisation of the Turbine Gland Seal

The TGS gland steam exhauster discharges gaseous radioactive waste into the main stack where it is combined with the OG and HVAC extract before being discharged into the environment. In 1976 a design change was introduced to prevent the leakage of reactor steam from the turbine into the turbine building and to avoid discharging unabated reactor steam directly into the environment. A diagram of the original design and the new design comprising a multi stage seal and separate steam seal supply system is show in Figure 5.1.10.9-2. The new design is more robust at preventing reactor steam leakage and is designed to return reactor steam to the main condenser rather than discharging part of it unabated to the environment.

The SSSS prevents the in leakage of air into the seal and subsequently into the main condenser. Air entering the main condenser would reduce the condenser vacuum pressure leading to a turbine trip (the current design would not function without the design modification that introduced the SSSS). Air that leaks into the main condenser also has the potential to become activated in the reactor increasing the generation of radioactive waste that would require management in the OG.

The SSSS uses steam created by the TGS. The TGS extracts water from the CST and converts it to steam using reactor steam in the turbine gland evaporator. Part of the turbine gland steam used in the turbine gland is then extracted with reactor steam into the main condenser. The remainder is extracted with any air in leakage to the gland steam condenser. 98 percent of the turbine gland steam is condensed in the main condenser and gland steam condenser and is subsequently reused in the plant.

The remaining residual steam (steam not condensed in either the main condenser or GSC) is discharged via the gland steam exhauster to the main stack. The discharge contains tritium and very small quantities of iodine and particulates and has been conservatively calculated to contribute approximately 2 percent of the dose from the UK ABWR. A HEPA filter is provided on the extract from the gland steam exhauster to remove the particulate matter. Condensing 98 percent of the steam in the main condenser and GSC also prevents the discharge of 98 percent of the tritium and iodine as it is entrained in the TGS steam.

The new design minimises the generation of radioactive waste compared to the original BWR design. It is recognised that the design does result in the discharge of gaseous radioactive waste. The abatement of the CST water prior to use and miniming the amount of residual steam that is discharged by condensing it results in the contribution to dose being very small (5.2.5.2. Evidence: TGS Impact of Gaseous Discharges).

As a result of this contribution to the gaseous radioactive waste discharges from the UK ABWR an assessment of options was undertaken to determine if the TGS could be further optimised. A robust process was implemented to both identify and assess the options (5.2.5.2. Evidence: TGS Options Assessment). During the assessment it was identified that to either reduce the source term of the steam or to reduce the radioactivity of the gaseous radioactive waste would result in significant cost (in terms of time, trouble and effort). A number of options also resulted in the generation of a large volume of aqueous radioactive waste that would require discharge (weakening argument 1h Recycling of water to prevent discharges).

Improvements in the TGS design since the 1970's have improved the leak tightness of the plant. Reactor steam is prevented from leaking into the turbine building where it has the potential to result in a discharge to the environment. The leakage of air into the main condenser is also prevented (5.1.10.10. Evidence: Improvements in turbine gland seal design). The combined performance improvements delivered by the TGS and TG in terms of minimising the radioactivity of gaseous radioactive waste, preventing the leakage of reactor water/ stream into the turbine building, minimising the in-leakage of air into the main condenser whilst preventing the generation of additional volumes of aqueous radioactive waste is considered to represent a significant benefit beyond that achieved by the original design. The assessment that explored opportunities to further optimise the TGS concluded that the costs (in terms of time, trouble and cost) are grossly disproportionate compared to the benefits in terms of dose reduction. The benefits of the existing design in addition to the other improvements detailed within this demonstration of BAT report that reduce the radioactivity of the CST water contribute to the application of BAT.

5.2.5.1.Evidence: TGS Impact on Gaseous Discharges

The radioactivity within the residual steam has been calculated based on the source term within the CST. The extract from the gland steam exhauster is provided with a HEPA filter to remove any particulate matter prior to it being combined with the HVAC system at the stack. Although a HEPA filter is provided this will only remove particulate matter and as such it has been assumed that it will have less impact on gaseous radioactive waste discharges which will mainly consist of tritium and iodine.

The changes in the design of the TGS will have a positive impact on the OG system as the TGS and TG prevent the in-leakage of air into the main condenser. Air that enters the main condenser can become entrained within the condensate and subsequently activated within the reactor resulting in greater challenge to the OG system, therefore minimising the generation of activation products and reducing the demand on the OG system. The baseline TGS design is not connected to the HVAC or OG system. These discharges from these three systems are combined in the main stack.

Condensate from the main condenser is returned to the CST and as such is subject to the same treatment as other water that is reused within the plant.

It is acknowledged that the TGS would result in a separate waste stream for gaseous tritium and iodine which would be discharged via the main stack independently of the OG and HVAC systems. Neither iodine or tritium is abated via the TGS route. The additional discharge and the impact on dose is given in Table 5.2.5.1-1.

Table 5.2.5.1-1: Quantification of TGS discharge

	Discharge from OG system (Bq/y)1	Discharge from HVAC system (Bq/y)1	Discharge from TGS system (Bq/y)1	Dose from OG system (μSv/y)	Dose from HVAC system (μSv/y)	Dose from TGS system (μSv/y)
Noble gases (Kr, Xe, Ar)	1.8E+12	0.0E+00	0.0E+00	5.8E+00	0.0E+00	0.0E+00
Carbon-14	9.1E+11	0.0E+00	0.0E+00	6.2E+01	0.0E+00	0.0E+00
H-3	0.0E+00	1.3E+12	1.6E+12	0.0E+00	1.2E+00	1.5E+00
Iodine	0.0E+00	2.7E+08	4.8E+07	0.0E+00	8.6E-01	2.3E-02
Particulate	0.0E+00	4.0E+04	2.1E+05	0.0E+00	1.2E-04	4.7E-04
Total				6.8E+01	2.1E+00	1.5E+00
Contribution(%)				95.0	2.9	2.1

Note: The annual discharges from each discharge route (OG, HVAC, TGS) are set by considering expected maximum discharge of each discharge route. (i.e. OG, TGS: 12 month operation, HVAC: 11month operation +1 month outage).

The preliminary dose calculations are performed using Dose Per Unit Release (DPUR) described in the IRAT methodology, which was developed by the Environment Agency. This DPUR is the same as the coefficient used in Stage 1 of E8 Prospective Dose Modelling [Ref-42]. The coefficient in the dose evaluation is very conservative. For that reason, the full prospective dose results show in E8 will be lower than that calculated here using the Stage 1 methodology.

The HEPA filter decontamination factor of 99.9 percent has been used as a removal factor for particulate in the discharge assessment.

5.2.5.2. Evidence: TGS Options Assessment

An assessment was carried out by Hitachi-GE to systematically review the TGS to identify those options that have the potential to impact on the generation of radioactive waste or on-site and off-site dose resulting from the TGS. The discussions and outcomes of the workshop are documented in detail in the Turbine Gland Steam System: Demonstration of BAT document [Ref-103] and are summarised below.

The options considered in the assessment were:

1. Revert to original BWR design (i.e. go backwards because the new design was the wrong choice),
2. Replace CST with HS,
3. Replace CST with purified water,

[1] Calculated from discharge data taken from [Ref-42].

4. Further treatment of CST water,
5. Replace sealing steam with air/ nitrogen,
6. Use less steam for seal,
7. Connect TGS discharge to main condenser or replace GSC with main condenser,
8. Install drain separator (demister plate) at TGS discharge,
9. Larger GSC,
10. Connect TGS discharge to OG condenser inlet,
11. Install cooler condenser at TGS discharge,
12. Add abatement at TGS discharge, and
13. Existing baseline TGS system design (the do nothing option).

The options assessment identified that the performance of the options was a balance between reducing gaseous radioactive waste discharges and increasing costs (in terms of time trouble and effort). It is also recognised that the increase in worker dose, industrial safety and non-radioactive environmental impact effectively result in an increase or reduction in cost (in terms of time trouble and effort) as the expectation is that a reduction in performance would result in a requirement for additional costs to apply mitigation when implementing the option. It was also identified that operability and aqueous radioactive waste were key attributes in differentiating the performance of the different options.

The outcome of this assessment identified that the baseline design represents an evolution of the design utilised in the early 1970s and continues to provide benefits in terms of safety and environmental performance. The baseline design supports the arguments presented within the Demonstration of BAT report in terms of minimising liquid waste discharges, whilst preventing leakage of reactor water/ steam from the TG and air into the TG. The use of CST water to produce the TGS seal steam does result in a gaseous radioactive waste discharge although this is considered to be optimised compared to the design used before 1976. The CST water is also subject to abatement to remove particulates and soluble radionuclides prior to use.

A robust process has been implemented to both identify and assess options that have the potential to further optimise the performance of the TGS. This assessment has concluded that the contribution to dose from this system is already very small reflecting the fact that the design is already optimised and that the introduction of further modifications would result in high costs (in terms of time, trouble and effort). On this basis it is concluded that the baseline TGS system is BAT and that further optimisation of the design is grossly disproportionate compared to the benefit in terms of both onsite and offsite dose reduction [Ref-103].

5.2.6. Argument 2f: Configuration of Liquid Waste Management Systems

Significant efforts have been made to minimise the generation of liquid radioactive waste. However, the safe and efficient operation of the UK ABWR requires that small volumes of radioactive liquids are disposed of to the environment, after appropriate treatment.

The design of the UK ABWR includes a liquid effluent system that collects, conveys and discharges aqueous radioactive wastes (5.2.6.1: Evidence: Configuration of Liquid Waste Management System). Where practicable, aqueous radioactive waste is treated prior to its discharge to the environment. Treatment systems include filtration, demineralisation and evaporation processes to remove certain materials and radionuclides which are considered in separate Arguments (5.2.8 Argument 2h Demineralisers for Distillates from the High Chemical Impurities Waste Evaporator and 5.2.9 Argument 2i: Evaporation of High Chemical Impurities Waste. The design has been developed by adopting a series of design policies (5.2.6.2: Evidence: Design Policies for the Liquid Waste Treatment System) that describe principles relating to the minimisation, segregation, containment, treatment and discharge of radioactive liquid effluent.

There is a system of segregated drains (5.2.6.1: Evidence: Configuration of Liquid Waste Management System) that allow wastes with broadly similar characteristics to be collected separately prior to the application of any treatment. The four drainage systems are:

- LCW;
- high chemical impurities waste (HCW);
- laundry waste (including shower drain); and
- controlled area drain waste.

The segregated drainage system ensures that treatment techniques can be targeted on specific characteristics of the waste stream and enhances the overall performance of the effluent management system. Treated effluent that meets the criteria for re-use within the reactor circuit is sent to the CST from where it can be pumped back into the reactor cooling circuit (5.1.8 Argument 1h: Recycling of Water to Prevent Discharges). Design data demonstrates that the quantity of liquid radioactive waste that is disposed of to the environment each year from the HCW drain will be extremely low and that no liquid radioactive waste will be disposed of from the LCW drain. The volume of waste water discharged from the other two drainage systems each year will typically be higher but, due to the source of these wastes the radioactivity will be very low.

The LCW treatment system is designed to treat liquid waste to a condition that is suitable for re-use within the facility. Only in the event that the volume of liquids within the cooling circuit and pools exceeds the maximum working capacity, treated/conditioned HCW will be potentially discharged from the facility rather than being returned for re-use. Design data demonstrates that the quantity of aqueous radioactive waste disposed of to the environment via this route each year will be extremely low. The water within the cooling circuit will only be discharged to the environment at the end of the operational life of the facility following appropriate treatment and assurance monitoring (5.2.6.4 Evidence: Key Parameters and Water Balance). CAD will be routinely discharged to the environment following treatment and assurance monitoring but the volume and radioactivity of this is expected to be very low. Some of the radionuclides in the aqueous effluent such as tritium and carbon-14 do not undergo treatment in the liquid effluent treatment system and are discharged directly to the environment. The majority of these techniques (5.2.6.3 Evidence: Assessment of Liquid Treatment Techniques for Tritium) that could be used to treat these radionuclides have been shown to have installation and operating costs that are very high, and considered grossly disproportionate compared with the low impacts associated with discharges. Therefore tritium and carbon-14 are typically returned to the reactor system where an equilibrium concentration will be reached. Only a small proportion of liquid waste is disposed of to the environment during the operational phase of the NPP and this is only in the event that liquids within the cooling circuit and pools exceeds the maximum working capacity. Additionally, concentrations are low and significant challenge is associated with treating low concentrations of tritium and carbon-14 in liquid wastes. As such, disposal to the environment subject to reassurance monitoring is therefore considered to be the BAT.

The UK ABWR design provides sufficient space to allow a future operator to carry out in-process monitoring to confirm that what has been identified as BAT is performing as expected (5.2.6.5 Evidence: In-process Monitoring to Support Demonstrating the Application of BAT).

5.2.6.1. Evidence: Configuration of Liquid Waste Management System

The LWMS segregates and collects liquid waste generated which has the potential to become contaminated during normal operation of the NPP. Treatment of the segregated liquid waste is determined based on the properties of the waste. Typically treated liquids are reused after being treated by the LWMS. However, in some cases they may be discharged after reassurance monitoring has demonstrated that concentrations of radioactive substances are very low.

The treatment capacity and the configuration of the systems provided in the LWMS are designed to manage the maximum design basis volume of liquid waste.

The LWMS comprises of the following four systems [Ref-28]:

- **Dedicated treatment system for low chemical impurities waste**

  LCW originates from:

- equipment drains (which includes T/B LCW sump, CUW blow down, reactor well drain, R/B LCW sump etc.);
- equipment blow down water (waste generated only in the case of periodic inspection); and
- SFP, FPC and CUW.

The LCW consists of hollow fibre filters, for the removal of insolubles, demineralisers, for the removal of solubles, and sampling pools. Treated liquids are returned to the CST. The selection of these treatment techniques was based on a Hitachi-GE assessment as discussed in Section 5.1.8.4.

- **Dedicated treatment system for high chemical impurities waste**

HCW originates from:

- RW/B HCW sump;
- chemical drains; and
- equipment blow down water (waste generated only in the case of periodic inspection).

The HCW comprises an evaporator for removal of impurities and a demineraliser for removal of residual solubles. The selection of these treatment techniques was based on a Hitachi-GE assessment which is described in sections 5.2.8 and 5.2.9. Treated liquids are either transferred to the CST for reuse or in limited circumstances disposed of to the environment following reassurance monitoring.

- **LD waste and hot shower drain waste**

LD and hot shower drain waste originates from:

- cleaning of personal protective equipment (PPE); and
- change room showers located on the clean side of barrier monitoring and controls.

LD waste water streams contain detergent, suspended solids and organic material, as well as potentially low levels of radioactive crud. Hitachi-GE undertook an assessment [Ref-59] to compare the different treatment technologies available for the LD. To remove these impurities the treatment system comprises of collection tanks and filters (LD pre-filter, LD activated carbon adsorption tower and LD filter) which manage and treat the waste prior to disposal to the environment following reassurance monitoring.

At the time of this initial assessment, the assessment team did not have access to detailed OPEX from Japanese operational facilities. Their assessment was therefore based on the technologies, which were judged to be best practice at Japanese operational sites and for which there was also UK OPEX. Upon the availability of additional detailed OPEX, Hitachi-GE undertook a further options assessment [Ref-107] to investigate all credible liquid waste options, to identify good practice and to ensure the chosen options are underpinned by robust and transparent justifications.

The assessment compared filtration and activated carbon, and the use of an external laundry facility.

Consignment of laundry to an external laundry facility, was provisionally identified as the most preferred option. This was because the use of an external permitted nuclear laundry offers a robust, proven solution for managing launderable PPE that utilises existing designated washing machines and effluent treatment facilities. Such machines and facilities are fully capable of handling and managing both alpha and beta contamination. It was also noted that, in the event of an external facility not being suitable, the application of filters and activated carbon to the LD effluent system would be a highly viable alternative.

This assessment concluded that whilst the use of an external permitted nuclear laundry would offer a robust, proven solution for managing launderable PPE, it is not feasible to adopt this option given the requirement of GDA. On this basis it was decided that the preferred option is to employ a dedicated on-site laundry and to treat the resultant effluent from this facility using filters and activated carbon. This supports the conclusion of the initial assessment [Ref-59].

CAD

This stream comprises various liquid wastes generated by plant and equipment in the UK ABWR facility's controlled areas, which are not otherwise captured by the HCW and LCW. For example, liquid derived from local air-conditioning units located in the controlled area. The quantity of CAD generation depends on the temperature and the humidity in the building.

Hitachi-GE undertook an options assessment [Ref-107] to investigate all credible liquid waste options, to identify good practice and to ensure the chosen options are underpinned by robust and transparent justifications. This assessment concluded that due to the very low concentrations of radiological and chemical contamination expected within this waste stream, indiscriminate application of effluent treatment technologies are judged to convey no overall benefit to the environment. Additionally, the use of abatement equipment would result in economic cost and environmental harm (in the form of resource use and secondary waste). Employment of such equipment was therefore considered disproportionate to the risk posed by the untreated effluent. To this effect, sampling and direct discharge, with the option to divert to the HCW system in the event that the effluent does not meet the discharge criteria, was considered the only, and therefore preferred, option.

The LCW, the HCW and the CAD will be installed in the RW/B. The LD waste system will be installed in the S/B. Figure 5.2.6.1-1 shows the LWMS.

Further detail on the LWMS is provided in PCSR Chapter 18 [Ref-58].

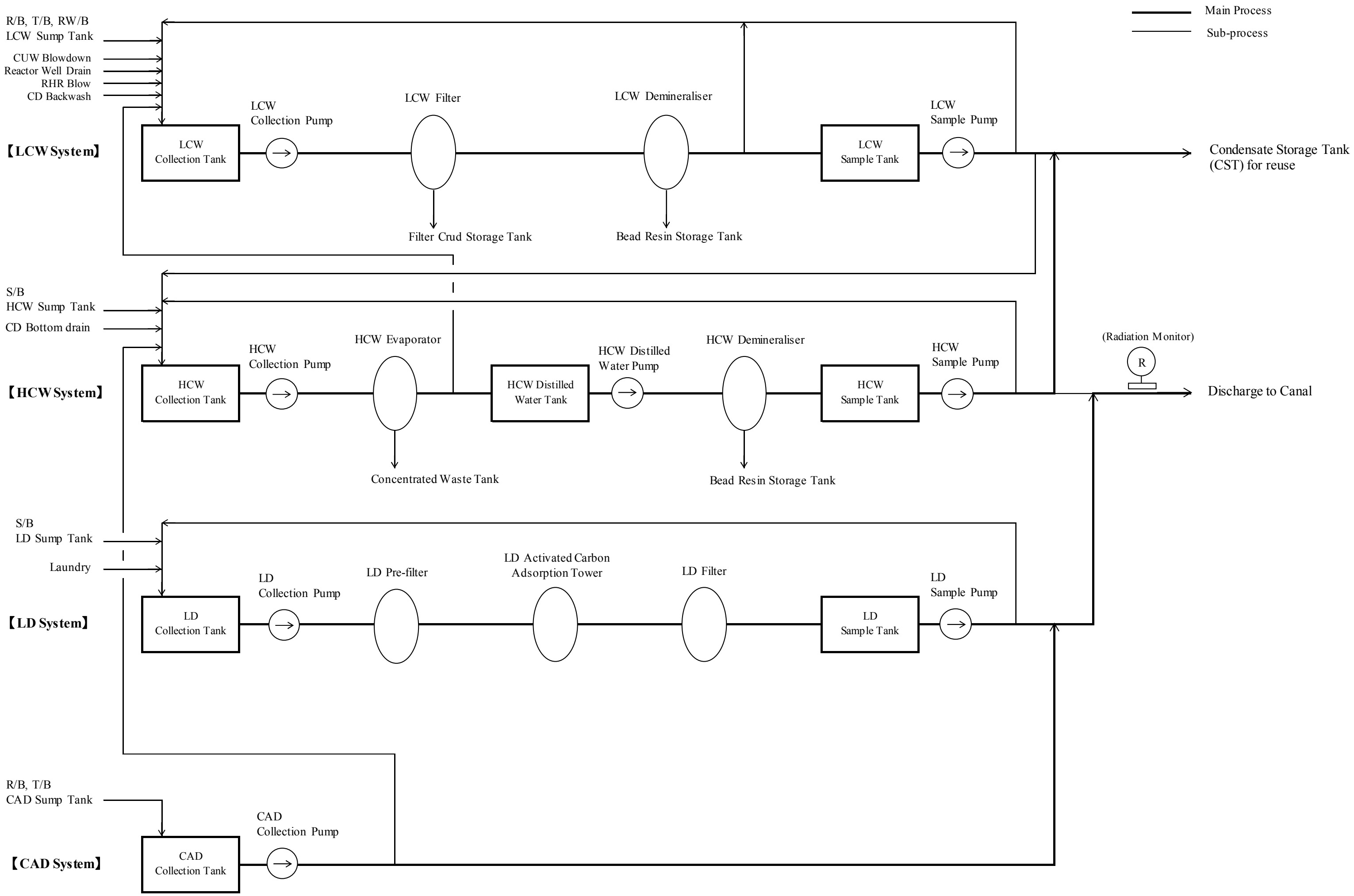


Figure 5.2.6.1-1: Liquid Waste Treatment System Flow Sheet

5.2.6.2. Evidence: Design Policies for the Liquid Waste Treatment System

The design of the LWMS have been developed based on the following design policies [Ref-28]:

1. The LWMS shall separate, collect and treat liquid wastes. As a general rule, the treated liquids shall be reused, and discharge of radioactive substances shall be minimised as far as practical.
2. The treatment capacity of the LWMS and the in-line configuration of the system shall be designed so that it will be able to deal adequately with anticipated cases in which the maximum amounts of waste liquids are generated. The components of the LWMS shall be made of suitable materials, taking into consideration the properties of the effluents.
3. The following items shall be taken into consideration in designing the facilities for treating liquid wastes and the facilities related to them in order to prevent leakage of liquid radioactive substances from these facilities and to prevent their uncontrolled discharge outside of the site:

    a. In order to prevent the occurrence of leaks, LWMS unit operations shall be made of suitable materials to prevent leaks and where appropriate will be provided with level alarms to aid detection of leaks and to prevent overfilling As a general rule, the drain pipes and vent pipes which empty outside of the system shall be provided with caps or similar closing devices. However, those which are used with a high frequency shall have drain vents leading to tanks or sump pits.
    b. Should radioactive liquids leak out, there shall be provisions making it possible to detect the leaks promptly and to remove and decontaminate the leaked liquids easily.
    c. The components of the LWMS either shall be provided inside an independent section, or bunds shall be provided around them in order to prevent spreading of any leaks inside the facilities. Weirs shall be provided on inlets and outlets connected to points outside the facilities in order to prevent leakage outside of the facilities. Outdoor devices and outdoor pipes shall be designed so that any leaked liquids will be collected inside facilities such as shielding walls or pipe ducts. Floor and wall surfaces of facilities in which there is a possibility that liquid radioactive wastes might collect will be constructed from materials that minimise the potential for leakage.
    d. Alarms for the tank water level or leakage detection shall be designed so that the output can be displayed either in the MCR or in the RW/B control room, so that it will be possible to inform the operators reliably of abnormalities, and so that they will be able to take suitable measures.
    e. The facilities shall be designed so that no floor surfaces within the facilities shall be located on drainage channels discharging effluents outside of the site in an uncontrolled manner. They shall also be designed so that no apertures connected to uncontrolled drainage channels shall be located inside the related facilities.

4. The LWMS shall be designed so that it can be centrally monitored and controlled in the RW/B control room. It shall also be possible to monitor this in the MCR.

A design review has been undertaken which has provided confirmation that the design policies have been appropriately addressed and implemented during the development of the UK ABWR design [Ref-56].

5.2.6.3. Evidence: Assessment of Liquid Treatment Techniques for Tritium

Given the design intent of the UK ABWR to recirculate all of the LCW and the majority of high chemical impurities waste within the NPP for the life time of the facility, discharges of tritium will be very low during normal operations.

Hitachi-GE has undertaken an assessment of techniques for the abatement of tritium in liquid wastes [Ref-67]. The assessment considered the following techniques:

- de-tritiation and tritium removal/separation;
- decay of liquids;
- evaporation; and
- conversion of tritiated water to solid waste.

Although the assessment identified that there are treatment processes which could, in theory, be used to reduce liquid tritium discharges from a UK ABWR, the cost and resources of their implementation is considered grossly disproportionate to the benefits in reduced public and environmental dose (which is very low) from the use of such processes. This position is strengthened by the fact that none of the treatment processes are currently used on operational ABWRs or other LWRs and would require significant development work to make them viable for operational use. There are currently no processes implemented on an industrial scale for the treatment of tritium in liquid phase for an ABWR.

By recycling and not making liquid discharges of reactor water; the majority of the tritiated water remains within the NPP for its lifetime. Some tritiated water will be lost via evaporation and when discharges are made from the HCW to maintain the water balance of the plant. The residence time of the reactor water within the plant for the lifetime of the NPP will contribute to minimising the radioactivity that becomes waste as a result of radioactive decay. The half-life of tritium is 12.3 years whilst most discharges will not occur until after 60-80 years allowing the radioactive decay of proportions of the tritiated water.

5.2.6.4. Evidence: Key Parameters and Water Balance

Table 5.2.6.4-1 provides a summary of the yearly estimated quantity of liquid waste generation from each system based on the cycle average (i.e. average of 17 month operation and 1 month outage) and the ratio that each liquid waste system contributes to the total volume of liquid waste.

Table 5.2.6.4-1: Estimated Quantities of Liquid Waste Being Treated

System	Volume (m^3/y) (cycle average)	Ratio to total
LCW	11,000	73.8%
HCW	400	2.7%
LD	1,900	12.8%
CAD	1,600	10.7%
Total	14,900	100%

Common practice for operators of ABWRs is to re-use treated LCW and HCW within the primary circuit. The LWMS is designed to process the liquid waste to meet the design criteria of treated water that can be returned to the CST for reuse [Ref-28], as shown in Table 5.2.6.4-2. If, after treatment, the LCW or HCW water quality does not meet the criteria for reuse (or discharge in certain circumstances for high chemical impurities waste), then the processed water is returned to the Collection Pool or Tank from where it originated via the re-processing line for re-treatment [Ref-52].

Table 5.2.6.4-2: Minimum Criteria to be achieved by Treatment to Allow Reuse

Item	Criteria
Conductivity	100 (μS/m)
Cl^-	20 (ppb)
pH	5.6 - 8.0
SO_4^{2-}	20 (ppb)
TOC	400 (ppb)

In some cases a future operator may be required to discharge a small proportion of the treated HCW to the environment, following reassurance monitoring, in order to maintain the water balance of the NPP. Detail on the operations that may result in the requirement to discharge surplus water is included in [Ref-57]. Examples include:

- water drained from the system during inspection of equipment; and
- activities during start up and shut down that result in the use of pure water such as using house steam as the driving force for the SJAE until switch over to the main steam.

Although only a proportion of the treated HCW will be discharged, if it is assumed that all treated HCW is discharged to the environment, the maximum volume of water from the HCW is 400m^3/year. This is approximately 4 percent of the total quantity of liquid waste (11,000m^3/year + 400m^3/year) generated from LCW and HCW. However, operational experience demonstrates that only a small number and volume of discharges from the HCW are made [Ref-48]. The approaches taken to minimise discharges from the HCW are [Ref-53]:

- Retaining and recycling water within the NPP. During operation, water is returned to the CST for reuse. During outages, several water tanks, including the CST are used to manage the storage of water during the plant outage. Upper pools can also be utilised as water storage during outage.
- Minimising the addition of treated fresh water to the system.

The Quantification of Discharges and Limits report [Ref-10] calculates the annual discharges from the HCW based on the assumption that all processed HCW is discharged to the environment.

All treated water from the LD and CAD system is discharged to the environment after reassurance monitoring to confirm that the water quality satisfies the discharge limits. Non-radioactive components including environmentally sensitive organic species are described in Section 8 of "Other Environmental Regulations" [Ref-61]. If LD processed water from the LD system cannot be discharged due to the water quality not satisfying the discharge criteria, the processed water is returned to the LD collection tank from the LD sample tank for further treatment. CAD effluent is collected in the CAD collection tank, and then sampled and analysed before being discharged to the environment. Usually CAD effluent is discharged to the environment without processing as it is not radiologically contaminated waste. If sampling and analysis identifies that the CAD water quality does not satisfy discharge criteria, the CAD effluent is transferred to the HCW collection tank for processing using the HCW [Ref-52].

The outline water balance for liquid wastes in the NPP is shown in Figure 5.2.6.4-1.

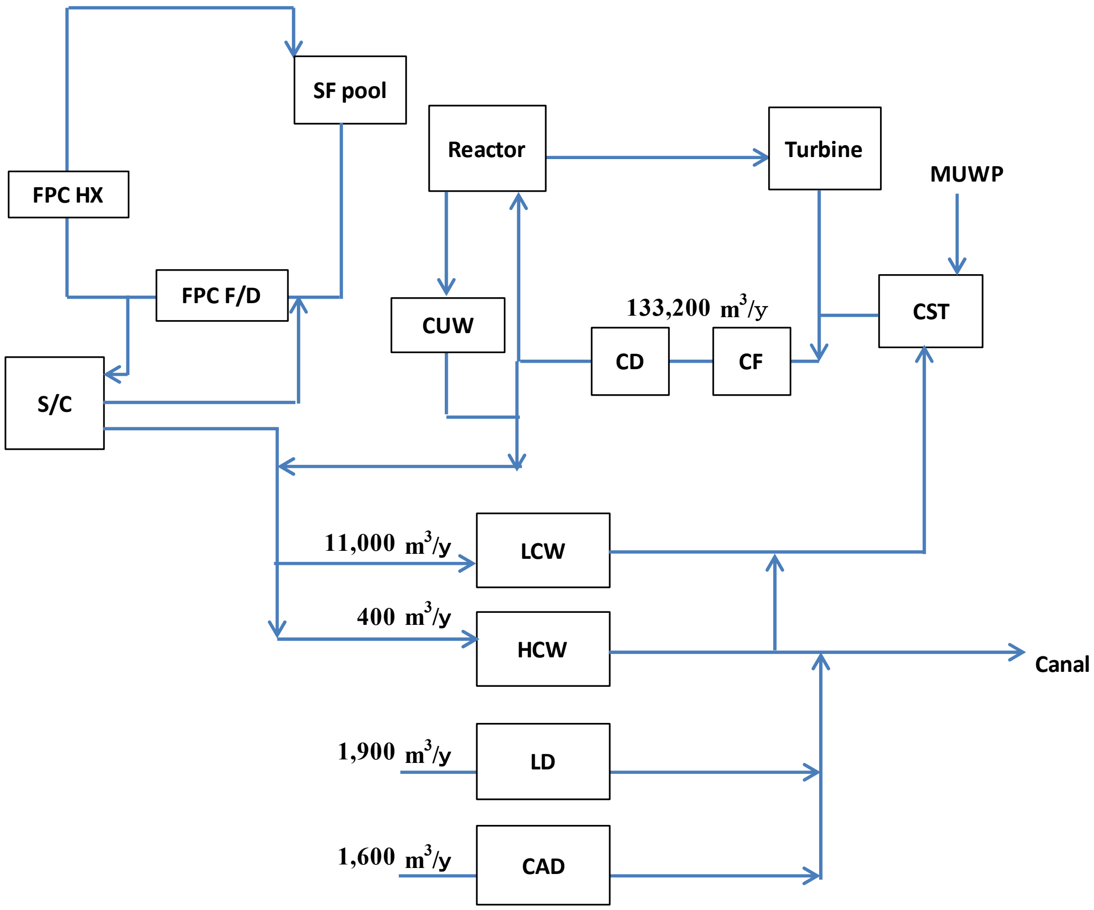


Figure 5.2.6.4-1: The Outline Water Balance of Liquid Wastes in the Plant

5.2.6.5. Evidence: In-process Monitoring to Support Demonstrating the Application of BAT

The components of the LWMS that require in-process monitoring are listed in Table 5.2.6.5-1 along with detail on the monitoring methods applied [Ref-20].

Table 5.2.6.5-1: Components of the LWMS Requiring In-process Monitoring and a Description of the Monitoring Techniques

Process system	Monitor	Location of monitoring	Objective	Type
LCW	Manual sampling	Manual sampling & analysis at LCW collection tank.	Analysing suspended solid and conductivity to confirm that the properties of collected liquid waste are suitable for the LCW.	Manual sampling & analysis.
	Conductivity meter	Inlet and outlet of LCW demineraliser.	Monitoring conductivity to confirm the processing performance of the demineraliser.	Continuous measurement.
	Manual sampling	Manual sampling & analysis at LCW sample tank.	Analysing to confirm that the properties of processed water satisfy the reuse criteria (Conductivity, pH, chloride ion (Cl^-), sulphate ion (SO_4^{2-}) and TOC).	Manual sampling & analysis.
HCW	Conductivity meter	Inlet and outlet of HCW demineraliser.	Monitoring conductivity to confirm the processing performance of the demineraliser.	Continuous measurement.
	Manual sampling	Manual sampling & analysis at HCW sample tank.	Analysing to confirm that the properties of processed water satisfy the reuse criteria (Conductivity, pH, chloride ion (Cl^-), sulphate ion (SO_4^{2-})) and TOC (or discharge criteria).	Manual sampling & analysis.
LD	Manual sampling	Manual sampling & analysis at LD sample tank.	Analysing to confirm that the properties of processed water satisfy the discharge criteria.	Manual sampling & analysis.
CAD	Manual sampling	Manual sampling & analysis at CAD collection tank.	Analysing to confirm that the properties of collected liquid waste satisfy the discharge criteria.	Manual sampling & analysis.

5.2.7. Argument 2g: Sizing of Tanks, Vessels and Liquid Containment Systems

The design of the UK ABWR includes a series of tanks to manage the liquid wastes from the various drain systems provided across the plant. These tanks have been designed (5.2.7.1: Evidence: Capacity of Tanks and Vessels) to provide the capacity necessary to store the effluent during treatment and prior to discharge. There are a separate series of tanks for each of the four dedicated treatment systems described in Section

5.2.6.1.

The tanks provide sufficient capacity to accumulate effluent from operational activities and expected events. The size of the tanks ensure that operators have enough time to undertake sampling and analysis of wastes prior to making any decisions to discharge effluent to the environment or to subject it to additional treatment. All tanks are fitted with a series of alarms that indicate when the tanks contain a pre-defined volume of liquid. In addition all tanks are contained in bunds to capture any spills from over-filling (5.2.7.2 Evidence: Secondary Containment of Tanks and Vessels). The tanks sizing, based on the calculations in 5.2.6.1, are therefore considered to be appropriate for normal operations and as such are considered to be BAT.

5.2.7.1. Evidence: Capacity of Tanks and Vessels

The tanks and vessels provided as part of the LWMS have been sized using design basis calculations to allow the collection, storage and recycling or discharge (depending on the particular system) of liquid effluents from the UK ABWR under normal operating conditions. The sizing of these tanks and vessels ensures that:

- capacity is sufficient to ensure that a future operator is not constrained by capacity when determining how to treat, re-use or dispose of liquid wastes; and
- the total site storage capacity of tanks and vessels is sufficient to retain estimated annual liquid volume production during normal operations (i.e. including volumes produced during shutdowns and other operational phases that generate effluent).

In early design phase, typically the capacity of the tanks and vessels is derived from which ever calculation [(1) or (2)] derives the greater capacity, where:

(1) is the normal quantity of waste generated over 2 days which can be stored in 1 tank; or

(2) is the maximum quantity of waste generated in 1 day which can be stored in 2 tanks.

$$(1)\quad \frac{\text{Normalquantity of waste a day m}^3 \times 2\text{ days}}{1\text{ tank}} \times 1.2 \times 1.1 = \text{Tank volume m}^3/\text{tank}$$

$$(2)\quad \frac{\text{Maximum quantity of waste a day m}^3 \times 1\text{ day}}{2\text{ tank}} \times 1.2 \times 1.1 = \text{Tank volume m}^3/\text{tank}$$

The calculated capacity of each tank and vessel is subsequently provided with a process margin (1.2) and free board margin (1.1) to ensure a conservative tank volume and to ensure that sufficient space is available within the tanks and vessels to allow for alarms and level detection as required. The capacity of the tanks is derived from the maximum volume that can generated by the feed process plus a process margin and a free board margin. The process margin ensures that the capacity of the tank is conservative and allows some flexibility to a future operator. The free board margin ensures detection systems and alarms can be appropriately installed and that they will allow sufficient response times for a future operator without a potential loss of containment. Further detail on the process margin and free board margin is provided in [Ref-47]. The liquid waste systems for each of the individual waste streams each have two sample tanks which allows one to be duty whilst the other is being sampled and then sentenced [Ref-58].

5.2.7.2. Evidence: Secondary Containment of Tanks and Vessels

Secondary containment will be provided for the tanks and vessels that form part of the LWMS. The secondary containment will be sized to ensure that 110 percent of the contents of the largest tank within the secondary containment can be retained. All containment surfaces will be constructed from materials that are impermeable to the wastes they are designed to contain.

5.2.8. Argument 2h: Demineralisers for Distillates from the High Chemical Impurities Waste Evaporator

As described in 5.2.6.4 the normal operating mode of the UK ABWR is to reuse treated HCW. However, when managing the water balance of the NPP an operator may be required to dispose of treated HCW to the environment. The evaporator in the HCW is effective at concentrating and containing the majority of the radioactivity from the HCW liquid into a form that will enable solid waste disposal and therefore minimise the radioactivity of the liquid waste that may potentially be discharged to the environment. However, some of the volatile radionuclides are carried over from the evaporator with the distillate and require treatment (polishing) to further minimise the radioactivity before the waste is potentially discharged to the environment.

The HCW includes a demineraliser, similar to those employed throughout the nuclear industry (5.1.8.5 Evidence: Nuclear Industry Application – Demineralisers), to remove soluble radionuclides from liquid processes and to allow the radioactivity to be managed as a solid waste.

Filtration is unnecessary as the evaporator retains solid matter in the concentrate. The demineraliser is capable of using a variety of resins (5.1.8.6 Evidence: Demineraliser Media) which allows the operator to make its selection based on operating requirements, compatibility with subsequent disposability requirements and any prevailing regulatory requirements. This flexibility is considered to represent BAT at the GDA stage.

The demineraliser will ensure that the concentration of FPs and activation products will be low in the event that the HCW is required to be disposed of to the environment. Reassurance monitoring will be undertaken to demonstrate that concentrations of radioactivity are below discharge limits prior to the waste being discharged to the environment.

5.2.8.1. Evidence: Configuration of the Demineraliser Provided in the HCW

The demineraliser on the HCW is used to remove soluble radionuclides from liquids and to allow the radioactivity to be managed as a solid waste. Details of the demineraliser are presented in Table 5.2.8.1-1.

Table 5.2.8.1-1: Description of the HCW Demineralisers

System	Number of Demineralisers	Type of Demineraliser
HCW	1	Mixed bed bead type ion exchanger

Following treatment by the evaporator, the radioactivity of distillate is low and as a result the radioactivity of the spent demineraliser resin is characterised as Low Level Waste (LLW). The radioactivity of the spent resin is shown in Table 5.2.8.1-2 below. The resin of the HCW demineraliser is exchanged when the conductivity at the outlet of the HCW demineraliser exceeds the CST collection criteria of 1μS/cm.

Table 5.2.8.1-2: Radioactivity of Spent Resins from the HCW

Source of spent resin	Radioactivity (Bq/cm^3)
HCW spent resin	4.0E+01

5.2.9. Argument 2i: Evaporation of High Chemical Impurities Waste

Liquid waste collected in the chemical drain contains substances with properties that interfere with waste treatment systems and can cause corrosion of process equipment. This waste is referred to as HCW and is segregated from the remainder of the process wastes which are referred to as LCW. The impurities in

HCW must be removed before the water can be treated by demineralisation and returned to the process or disposed of to the environment.

The design of the UK ABWR includes an evaporator to treat the HCW and to remove the impurities. The evaporator has been designed to ensure discharges are minimised as far as reasonably practicable and are standard components in ABWRs operating in Japan. This was subject to an options assessment which concluded that this method remains BAT(5.2.9.1: Evidence: Nuclear Industry Application - Evaporators) (5.2.9.2: Evidence: Configuration of Evaporation System). The design of the evaporator has undergone a number of improvements to account for the increase in demand resulting from inclusion of floor drains in the HCW stream (5.2.9.4: Evidence: Evaporation System - Design Improvements). These improvements have further reduced the corrosion potential of the re-circulated HCW and reduced the accumulation of scale that could impact on the performance of the evaporator.

Use of the evaporator allows a high proportion of the HCW to be returned to the process. All liquid waste from the evaporator is passed through demineralisers that remove any soluble/volatile radionuclides that are carried over. The residues from the evaporator which contain the majority of the radioactivity are converted to and disposed of as solid radioactive waste. The design intent of the UK ABWR is that a future operator will only discharge liquid waste to the environment from the HCW in order to maintain the water balance of the NPP. The low frequency of the discharges combined with the application of robust treatment technologies are considered to represent BAT. Sampling and monitoring is carried out within the HCW in order to demonstrate the performance of the evaporator and demineralisers and to ensure treatment meets the reuse or discharge criteria (5.2.9.3 Evidence: HCW sampling and monitoring).

5.2.9.1. Evidence: Nuclear Industry Application - Evaporators

Evaporators are used for the treatment of radioactive liquid effluents; they are widely used throughout the nuclear industry and are standard practice for BWRs, ABWRs and PWRs. The UK ABWR proposes to make use of evaporation to minimise the discharge of radioactive liquid effluents from the HCW. Evaporation of effluents results in the production of a sludge-like concentrate that will contain the bulk of the radioactivity from the HCW.

Evaporation leads to significant volume reductions compared with other techniques. Depending on the chemical composition of the liquid effluents and the evaporator type, a DF of between 1,000 and 10,000 is typically achieved. The decontaminated water is collected in the vapour phase whilst most of the radioactivity is retained as a solid (non-volatile components such as salts containing most radionuclides) at the bottom of the condenser.

The IAEA report on the Handling and Processing of Radioactive Waste from Nuclear Applications [Ref-29] describes how evaporation is widely used in the nuclear industry as an effective method for chemical and radiological purification of liquid effluent. An Oslo and Paris Convention on Protection of the Marine Environment of the North East Atlantic (OSPAR) report [Ref-30] found that amongst the Contracting Parties of the OSPAR agreement, evaporation is a widely used technique in the treatment of liquid discharges arising from nuclear power generation.

Hitachi-GE originally undertook a study to review the treatment technology provided to abate the HCW [Ref-59]. The study assessed a number of different treatment techniques; evaporation, reverse osmosis membrane, ion exchange, ultra filtration and micro filtration and gave each technique a score for a range of criteria including operational experience, reliability, maintainability, solid waste generation, DF and cost. Overall, evaporation scored highest in the study. The evaporation technique outperformed the other techniques in important areas including;

- the highest DF of 10^3 to 10^4;
- the lowest impact on solid waste generation due to the evaporator's high volume reduction performance; and
- reliability safety due to the evaporator's extensive operating performance.

The evaporation technique performed less well in terms of capital cost and layout impact (size of installation) however these were deemed less significant compared to the main benefits identified above.

A second options assessment was undertaken to determine whether there is an option that performs better, over a range of relevant criteria, than the evaporator which is currently included in the design. A multi-criteria analysis methodology was used to undertake the second assessment [Ref-105]. The study identified that based on the waste inventory available at GDA only evaporation and reverse osmosis were capable of meeting the stringent purity requirements of process water in the UK-ABWR. The study also identified that there was little to discriminate between these two options and that neither presented any novel challenges in terms of safety, environmental performance, technical performance, cost or interfaces with other elements of the plant. On this basis, it was concluded that there are no grounds for changing the design of the HCW and that the evaporator should remain.

5.2.9.2. Evidence: Configuration of Evaporation System

The evaporator is part of the HCW and receives waste liquid from the HCW collection tank which has undergone pH neutralisation. The forced circulation evaporators provided within the UK ABWR design have appropriate capacity. A diagram of the HCW including the evaporator units is provided in Figure 5.2.9.2-1. Following treatment in the evaporator the treated liquid is collected in the HCW distillate tank before further treatment in the HCW demineraliser.

The concentrated HCW from the bottom drain of the evaporator will be transferred into the concentrated waste tank for a defined period to decay short half-life radionuclides and then the concentrated HCW will be transferred for treatment in accordance with the waste treatment BAT options assessment study [Ref-43]. The properties of the evaporator concentrate residue are described in the Radioactive Waste Management Arrangements document [Ref-36]. A study was conducted to identify the optimal conditioning and disposal route [Ref-43] for this waste stream. Further detail on this is provided in Section 5.3.4.1.

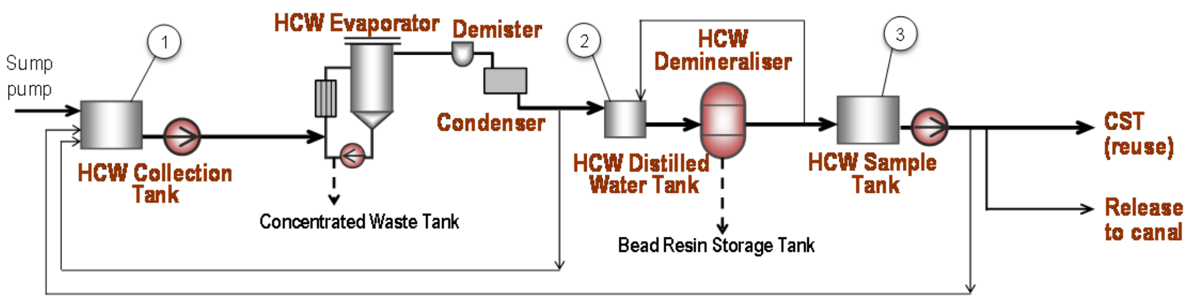


Figure 5.2.9.2-1: The HCW Sampling Points

Table 5.2.9.2-1 below shows the level of decontamination achieved across the HCW [Ref-53].

Table 5.2.9.2-1: Activity across the HCW (Source Term Topic Report [Ref-44])

Nuclides	HCW Collection Tank (Bq/cm^3)	HCW Distilled Water Tank (Bq/cm^3)	HCW Sample Tank (Bq/cm^3)
FPs and CPs (excluding H-3)	4.4E+01	4.4E-01	4.4E-02

5.2.9.3. Evidence: HCW Sampling and Monitoring

The HCW includes a number of sampling and monitoring points to confirm the performance of the system and to ensure the criteria for reuse or discharge is met [Ref-62]. The locations of these monitoring points are shown in Figure 5.2.9.3-1. The purpose of the sampling points is described in Table 5.2.9.3-1; further detail of sampling can be found in the approach to sampling and monitoring report [Ref-20].

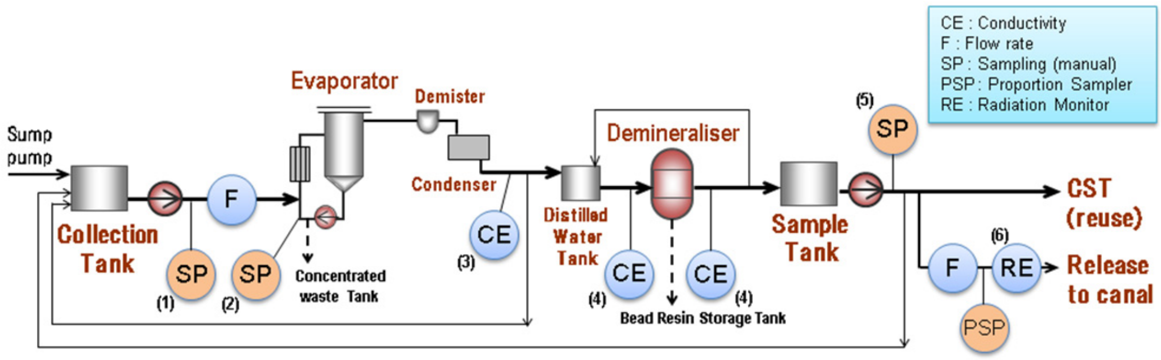


Figure 5.2.9.3-1: Outline of HCW and Monitoring/Sampling Points

Table 5.2.9.3-1: Items and Purpose for Each Monitoring and Sampling Point

Monitoring/ Sampling No	Measured Items	Purpose of Measurement
(1)	Conductivity Chloride ion Suspended solid Radioactivity pH, etc.	(Manual sampling & analysis) To confirm the properties of liquid waste supplied to the evaporator.
(2)	Suspended solid Chloride ion, etc.	(Manual sampling & analysis) To manage the concentration of concentrated liquid waste in the evaporator, these items are measured periodically.
(3)	Conductivity	Monitoring of the carryover from the evaporator to confirm whether resultant liquid waste (distilled water) is suitable for demineralisation process.
(4)	Conductivity (Inlet/Outlet)	To confirm the processing performance of demineraliser.
(5)	Reuse criteria or Discharge criteria	(Manual sampling & analysis) To confirm that the properties of processed water satisfy reuse criteria (or discharge criteria).
(6)	Radiation	During the discharge, radiation levels are continuously monitored and if the monitor signal exceeds the pre-set value, discharge automatically stops. (*1) (*1) Before the discharge, manual sampling and activity check is performed at the sampling tank.

5.2.9.4. Evidence: Evaporation System - Design Improvements

A number of design improvements have been introduced into the design of the ABWR that have resulted in a reduction in the amount of radioactivity that is discharged into the environment. The improvements include:

- **Volume reduction of floor drain waste (reuse of distillate water) by evaporation** [Ref-28]

  In early generations of BWRs floor drain waste was disposed of to the environment following filtration. This resulted in the dominant discharge of radioactivity to the environment originating from the floor drains (approximately 90%). Design improvements have resulted in the floor drain being treated by the HCW where evaporation and demineralisation processes are applied to treat it. This has resulted in a number of benefits which include a significant reduction in the radioactivity discharged to the environment and an increase in the volume of water that is available for reuse within the NPP. Treatment and reuse of the floor drain resulted in an approximate reduction in radioactive discharges to 1/10 [Ref-13].

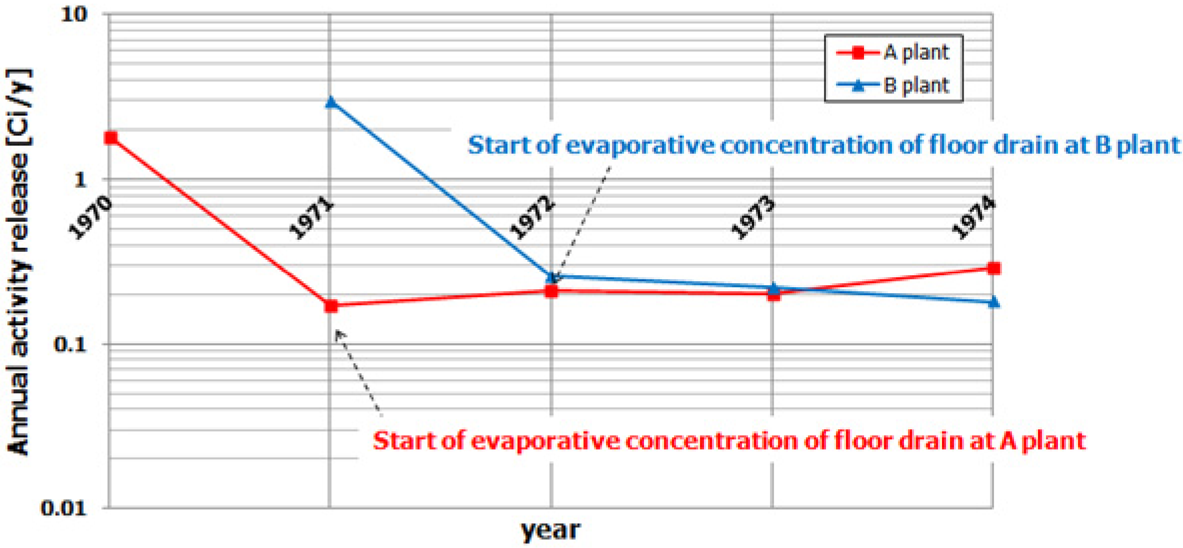


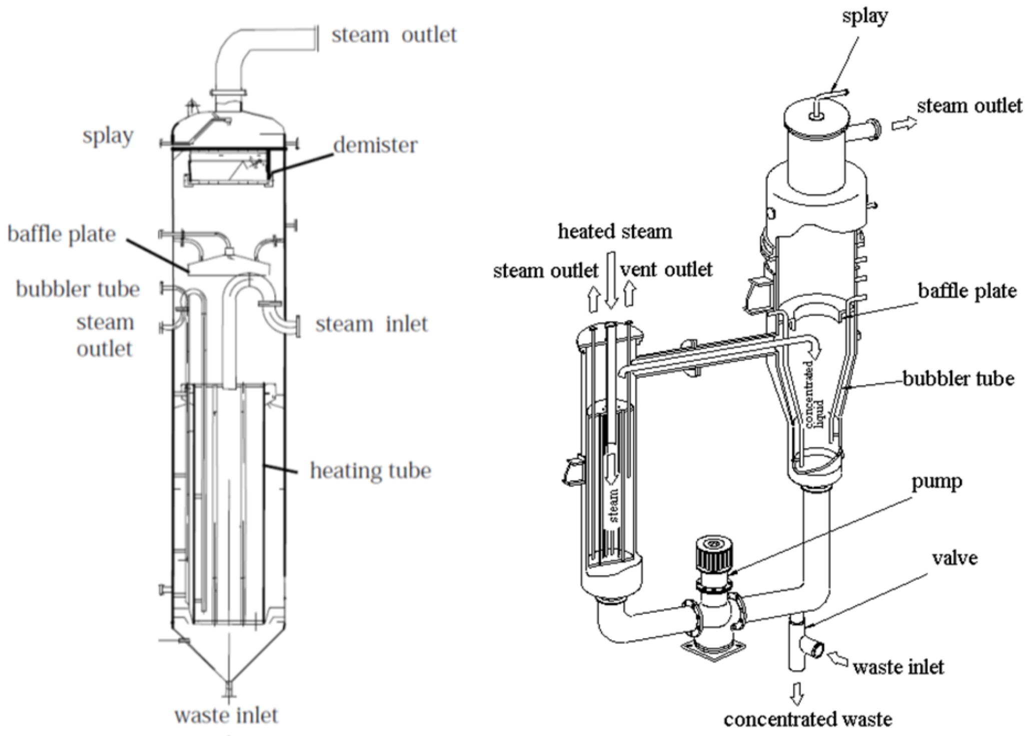
Figure 5.2.9.4-1: Released Radioactivity (except tritium) from Liquid Waste Treatment System to Environment (A Plant and B plant in JAPAN)

- **Adoption of a forced circulation evaporator**

As a result of combining the floor drain waste and HCW, a more durable and reliable evaporator was required to meet the requirements of the combined waste stream. The original evaporation system was a single barrel or natural circulation multi-barrel type. This was not sufficiently robust due to the:

  - occurrence of pitting corrosion and crevice corrosion; and
  - occurrence of blocking scale in the heat exchanger tube.

To prevent blocking by scale, a forced circulation (by pump) evaporation system was developed by Hitachi-GE. This type of evaporator increases the velocity in the heat exchanger which reduces the occurrence of blocking scale. This improved evaporation system has been adopted in the Standard ABWR design and will be provided within the UK ABWR. The design changes that lead to the improvements stated above are illustrated in Figure 5.2.9.4-2 below.



Natural Circulation Type Evaporator   Forced Circulation Type Evaporator

Figure 5.2.9.4-2: Illustration of Evaporator Design Improvements

- **pH Adjustment**.
  The capability to adjust the pH of waste entering the evaporator has been provided within the ABWR design; this system enables the corrosion potential of the waste to be reduced.

5.2.10.Argument 2j: Radioactive Decay of Solid and Liquid Wastes

Over time the amount of radioactivity associated with all radioactive wastes will reduce as a result of radioactive decay. The rate at which the radioactivity reduces depends on the 'half-life' which is different for each of the radionuclides present within the waste streams. Allowing solid and liquid radioactive waste to undergo radioactive decay before disposing of it to the environment or another premises will reduce the amount of radioactivity that is disposed of in the waste. The reduction of radioactivity will be a function of the half-life of the radionuclides and the length of time over which the waste is stored (5.2.10.1 Evidence: Nuclear Industry Application – Decay Storage).

The design of the UK ABWR includes a number of features that allow solid and liquid radioactive wastes to undergo decay prior to disposal. In some instances these features are dictated by factors that are beyond the control of the owner/ operator such as the availability of a waste management solution for HAWs (5.2.9.2 Evidence: Storage of Solid ILW). In other instances, such as the frequency of running batch treatment plants for processing waste prior to disposal (5.2.9.3 Evidence: Radioactive Decay of Sludge Waste, 5.2.10.4 Evidence: Storage Period of CUW and FPC Resins and 5.2.10.5 Evidence: Decay Storage of Concentrated Liquid Waste), the owner/operator will be able to take account of radioactive decay in its decision making process.

During 60 years of operation the RPV and a number of RIN become activated and are expected to comprise LLW, ILW and HLW at arising. Characterisation and segregation will be applied as far as practicable, subject to future operator BAT and ALARP assessment, to the waste as it arises.. The majority of the RIN inventory is estimated to be ILW at arising. Utilisation of radioactive decay is applied to ILW during decommissioning in order to reduce its radioactivity (5.2.10.6 Evidence: Decay storage of ILW during decommissioning). The period of time that an operator stores a waste and the decision to apply pre-treatment techniques prior to storage will need to be balanced against the key attributes such as the requirement to reduce the hazard presented by the form of the waste. Any decay that can be achieved prior to disposal will contribute to reducing the radioactivity that requires disposal and as such contributes to the application of BAT.

The higher activity waste tanks used for the decay storage of this waste will be fitted with a Tank Vent Treatment System (TVTS). The Rw/B TVTS is connected to the ullage space of the most active tanks within the Rw/B with the purpose of maintaining a depression, so that the air displaced due to effluent transfers is drawn into the TVTS and to provide dilution of potentially hazardous gases which arise within certain areas of the process by providing a constant air flow through the tank ullage space. Exhaust air is provided with primary and secondary HEPA filtration to remove any airborne particulates. Each stage of filtration is also provided with duty and standby filters to enable filter change operations to be undertaken whilst the system remains operational. Air is discharged via a discharge duct, downstream of the building HVAC HEPA filters, into the base of the main exhaust stack (5.2.10.7 Evidence: Tank Vent Treatment System).

5.2.10.1.Evidence: Nuclear Industry Application - Decay Storage

Decay storage is a recognised practice in the nuclear industry. The OECD report on Effluent Release Options from Nuclear Installations [Ref-21] states that wastes should be capable of safe storage prior to final disposal in a repository in support of optimisation of waste disposals.

The Department for Environment, Food and Rural Affairs (DEFRA) Radioactive Waste Management Advisory Committee [Ref-31] also stated that safe storage means the safe containment and storage of a package of waste, possibly for several decades, before its final disposal in a repository.

The Nuclear Industry Safety Directors Forum, in its report on BAT for the Management of the Generation and Disposal of Radioactive Wastes [Ref-32], detailed how the adoption of the waste hierarchy is embedded in UK policy for the management of solid, liquid or gaseous radioactive wastes. One of the principles of the waste hierarchy, in relation to decay storage, is minimising quantities of waste requiring disposal by adoption of decay storage.

Environment Agency guidance [Ref-23] states that for “nuclides with short half-lives that decay to stable (or less hazardous) nuclides, storage prior to discharge represents an option for abatement”. The process of decay storage reduces the activity and therefore minimises the amount of radioactivity discharged to the environment. The guidance also noted that “For radioactive decay to reduce the inventory of any nuclide to 10% of its initial value requires a storage period of between three and four half-lives. This delay option is therefore viable only for shorter lived nuclides”.

5.2.10.2.Evidence: Storage of Solid ILW

It has been assumed that solid ILW will be stored initially in the SFP for a minimum of 5 years following removal from the reactor due to the thermal loading of the waste and the campaign cost benefit requirements.

Following this initial storage period, solid ILW will then be placed into a buffer store to allow further radioactive decay prior to size reduction (where necessary) and packaging in an appropriate container, pending final GDF disposal.

5.2.10.3.Evidence: Radioactive Decay of Sludge and Spent Resin Waste

The waste sludge from the filters of the LWMS is stored temporarily in the settling and separating tanks of the spent sludge system. The settling and separating tanks have a storage capacity exceeding the amount calculated to be generated in up to approximately 5 years. However, a future operator will determine the precise storage capacity of tanks taking account of operability, hazard reduction, public dose and environmental impact. Additional information on this can be found in Chapter 18: Radioactive Waste management, of the PCSR [Ref-59].

The spent resins generated by the CDs of the Condensate Clean-up System and by the demineralisers of the LWMS are stored in spent resin tanks which have a storage capacity exceeding the amounts likely to be generated in 5 years.

5.2.10.4.Evidence: Storage Period of CUW and FPC Resins

Table 5.2.10.4-1 shows the relationship between the decay period and the estimated radioactivity concentration for spent CUW and FPC resins; it indicates that decay storage of CUW/FPC resins reduces radioactivity to approximately 1/100 over a period of 30 years. After which the rate of decay significantly reduces. The UK ABWR design includes the capability to immobilise resins into a passively safe state. Taking this into consideration, the decay storage of resins in the UK will be undertaken after immobilisation of the resins.

Table 5.2.10.4-1: Decay Period and Activity of CUW and FPC Resins

	0 year	**10 years**	**20 years**	**30 years**	**40 years**	**50 years**	**60 years**	**70 years**	**80 years**	**90 years**	**100 years**
CUW resin total activity (Bq/g)	1.3 E+08	1.5 E+07	4.0 E+06	1.9 E+06	1.3 E+06	1.1 E+06	9.5 E+05	8.6 E+05	7.9 E+05	7.2 E+05	6.7 E+05
FPC resin total activity (Bq/g)	1.7 E+07	1.9 E+06	5.0 E+05	2.2 E+05	1.5 E+05	1.2 E+05	1.1 E+05	1.0 E+05	9.4 E+04	8.7 E+04	8.2 E+04

5.2.10.5.Evidence: Decay Storage of Concentrated Liquid Waste

The concentrated liquid waste from the evaporator in the HCW is collected in concentrated waste tanks and can be stored for one year to reduce the radioactivity by decay. After this period of decay, the concentrated waste is mixed with a solidifying agent and placed into drums.

Halogenous nuclides are expected to reduce significantly in radioactivity over the decay period of several months. FPs, other than halogenous nuclides, have much longer half-lives and as such will not significantly decay over short time periods (e.g. months or even years).

Taking into consideration the above, the storage period in the tank of the concentrated waste has been determined as one year to allow halogenous nuclides to decay. The relationship between the decay period and the radioactivity concentration in the concentrated waste tank based on the Japanese BWR plant is shown in Figure 5.2.10.5-1. Actual precise storage capacity will be decided during detailed design phase.

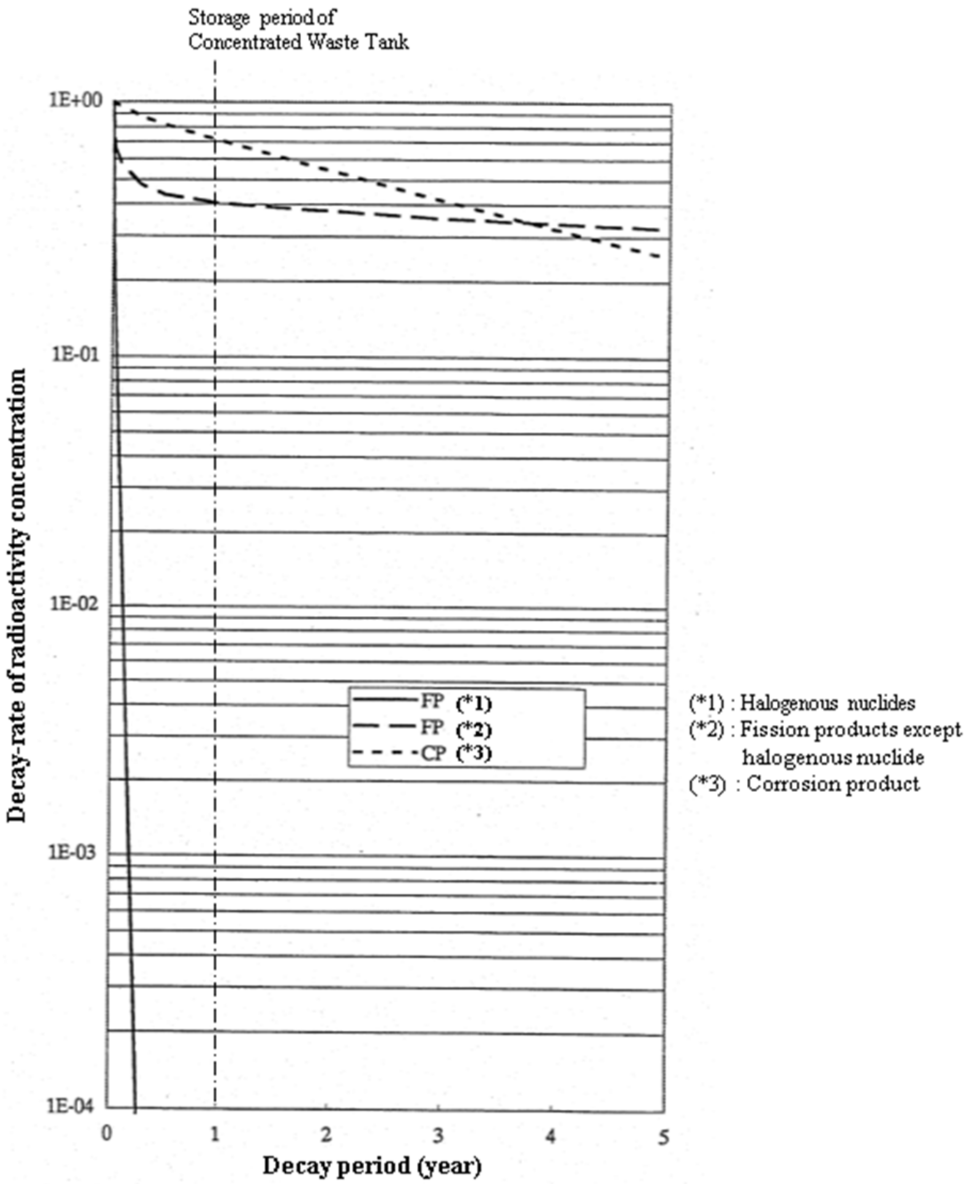


Figure 5.2.10.5-1: Relationship between Decay Period and Radioactivity

5.2.10.6.Evidence: Decay storage of ILW during decommissioning

As the UK ABWR enters into decommissioning, SF is stored in the SFP within the R/B for a period of up to 10 years before it is transferred to the SF storage facility. Following the removal of SF from the R/B, RINs can be dismantled and are treated as solid ILW (including a part of the HLW) which is stored in the additional ILW store (or HLW decay storage facility) for approximately 50 years before transfer to the Geological Disposal Facility (GDF) for final disposal [Ref-100][Ref-106]. If the waste is stored for 50 years a reduction in radioactivity of 79 percent would be achieved as shown in Table 5.2.10.6-1. RINs generated as HLW (70 tonnes within 220 tonnes) will be decayed to ILW and disposed of to the GDF. The RPV and the RPV thermal insulation will also be disposed of as LLW following decay storage.

Table 5.2.10.6-1: Decay in storage of ILW solid waste

Products		Weight (tonnes)	Activation Radioactivity (Bq/tonne)	
			6 years	50 years
RINs	Core shroud, Top guide, Core plate, etc.	220	3.5E+15	7.4E+14
RPV	RPV, RPV cladding	580	1.8E+10	9.0E+8
Others	RPV thermal insulation (Core region)	10	1.9E+10	1.1E+9
Total		810	9.4E+14	2.0E+14
Reduction rate			(100%)	79%

5.2.10.7.Evidence: Tank Vent Treatment System

The higher activity waste tanks used for the decay storage of this waste will be fitted with a Tank Vent Treatment System (TVTS). The Rw/B TVTS is connected to the ullage space of the most active tanks within the Rw/B with the purpose of maintaining a depression, so that the air displaced due to effluent transfers is drawn into the TVTS and to provide dilution of potentially hazardous gases which arise within certain areas of the process by providing a constant air flow through the tank ullage space.

The higher activity tanks selected for connection to the Tank Vent Treatment system are associated with following systems:

- Spent Resin & Sludge (SS) Storage.
  - Powder Resin Storage Tanks.
  - Bead Resin Storage Tanks.
  - Filter Crud Storage Tanks.
- Wet-solid ILW (WILW) Process Tank.
- Wet-solid LLW (WLLW) Process Tank.

Exhaust air is provided with primary and secondary HEPA filtration to remove any airborne particulates.

Each stage of filtration is also provided with duty and standby filters to enable filter change operations to be undertaken whilst the system remains operational. Air is discharged via a discharge duct, downstream of the building HVAC HEPA filters, into the base of the main exhaust stack. A schematic diagram of the Tank Vent Treatment system is provided in Figure 5.2.10.7-1 below which shows the tanks to which the Tank Vent Treatment system is connected, hot air in-bleed and filter arrangements.

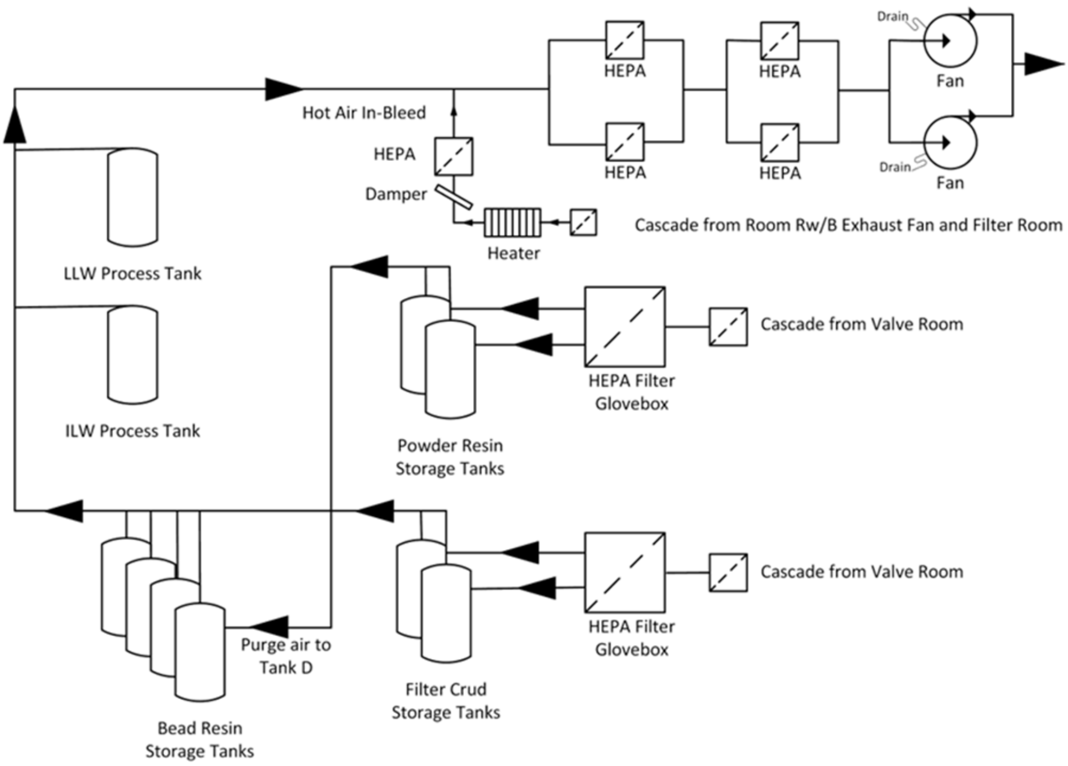


Figure 5.2.10.7-1: Schematic of the Tank Vent Treatment System

The creation, management and disposal of radioactive waste has been optimised through the use of Best Available Techniques (BAT). This is achieved by:

- The design of the Tank Vent Treatment system follows established codes of practice and relevant good practice;
- Discharge filtration is provided by two-stage duty standby HEPA Filters;
- HEPA filters on ventilation air inlets prevent dust entering the ventilation system, reducing the dust burden on the discharge HEPA filters, thereby extending their life;
- Hot air in-bleed system prevents discharge HEPA filters being exposed to high humidity, which maximises the HEPA filter lifetime, keeping changes to a minimum; and
- HEPA filter performance and flow parameters are monitored and alarm if any abnormal flow state exists or operational parameters are compromised.

5.3. Claim 3 - Minimise the Volume of Radioactive Waste Disposed of to Other Premises

The Arguments presented in support of this Claim are considered to demonstrate compliance with the standard BAT conditions [Ref-2]:

- Condition 2.3.2(b) 'The operator shall use the best available techniques in respect of the disposal of radioactive waste pursuant to the permit to minimise the volume of radioactive waste disposed of by transfer to other premises.'

This is also considered to fulfil the following requirement of the P&ID [Ref-1]:

- Minimising (in terms of mass/volume) solid and non- aqueous liquid radioactive wastes and SF.

The UK ABWR design contains a range of features that contribute to the substantiation of this Claim including:

- Design changes that will minimise the volumes of operational and decommissioning waste arisings.
- Provision of a number of features that will allow future operators to adopt an operating philosophy that will minimise the quantity of solid radioactive waste associated with routine operations and maintenance.
- Provision of dedicated facilities for the management, treatment and storage of solid radioactive waste.
- Reducing the quantity of solidified HCW that are generated.
- Availability of a range of decontamination techniques during decommissioning.
- The use of the clearance and exemption process to remove wastes from radioactive waste controls.

In developing the Arguments presented to demonstrate the validity of Claim 3 the REPs have been taken into account. The following REP is considered to be specifically relevant to this Claim:

Principle RSMDP3 'the best available techniques should be used to ensure that production of radioactive waste is prevented and where that is not practicable minimised with regard to activity and quantity.'

5.3.1.Argument 3a: Design to Minimise the Volumes of Operational and Decommissioning Waste Arisings

The operation, maintenance and subsequent decommissioning of the UK ABWR will generate solid radioactive waste that will require management and treatment before it is consigned for either disposal to other premises or storage on-site pending future disposal. The design of the UK ABWR has evolved to reduce the quantities of solid radioactive waste that will be generated during its life-cycle and to ensure that those wastes that are unavoidably created are compatible with waste management techniques typically used in the UK. The design changes that have had the greatest impact on the volume of solid waste generated are:

- RIPs are used to circulate the water coolant around the reactor in the UK ABWR. These internal pumps have removed the pipe work and associated plant used in previous evolutions to pump the water coolant and will contribute to a reduction in the quantity of waste generated from maintenance and decommissioning operations. RIPs also have the added benefits of reducing occupational exposure to workers, decreasing the amount of power required to recirculate the water coolant and reducing the cost of vessels and pipe work (5.3.1.1. Evidence: Introduction of Reactor Internal Pumps) compared to the previous design.
- Evolution of the BWR design replacing the PCV comprising a free standing steel containment vessel (SCV) and separate Biological Shield Wall (BSW) with the ABWRs Reinforced Concrete Containment Vessel (RCCV) with integrated steel liner. This design improvement delivers the same functionality whilst reducing the volume of metal and concrete that will require disposal at decommissioning (5.3.1.2 Evidence: Evolution of the Primary Containment Vessel).
- The UK ABWRs PCV is smaller than those used in previous evolutions of the BWR. This has the effect of reducing the size of the Radiological Controlled Area (RCA) by approximately 15%

which will reduce the quantity of waste generated from general occupancy, housekeeping and decommissioning (5.3.1.3 Evidence: Design of the Primary Containment Vessel).

- Introduction of techniques that reduce the amount of SCC experienced on reactor components. These techniques contribute to a reduction in the frequency that components require replacement as a result of corrosion. Consequently, this will result in a reduction in the quantity of waste associated with the replacement of damaged components and any related maintenance activities (5.3.1.4 Evidence: Techniques to Reduce Stress Corrosion Cracking).
- The introduction of Hollow Fibre Filter (HFF) or pleated filters that have eliminated the powder resins that were generated on previous evolutions of the BWR from the use of pre-coat filters. The use of HFF's or pleated filters is expected to reduce the quantity of solid radioactive waste by 27 tonnes per year (5.3.1.5 Evidence: Replacement of Pre-coated Filters).
- The introduction of a range of design changes that contribute to a reduction in the volume of operational and decommissioning waste (5.3.1.6 Evidence: Review of Further Design Changes).

The above improvements have been developed and implemented as a result of Hitachi-GE's on-going commitment to improving performance. Hitachi-GE has a comprehensive research and development programme that explores opportunities to reduce the materials used during construction and operations.

5.3.1.1. Evidence: Introduction of Reactor Internal Pumps

The method of re-circulating water in the reactor has undergone a number of evolutionary changes since the BWR concept was originally conceived. Figure 5.3.1.1-1 provides a summary of the evolutionary steps [Ref-33], which have progressively reduced the amount of process equipment, pipework and materials associated with the reactor.

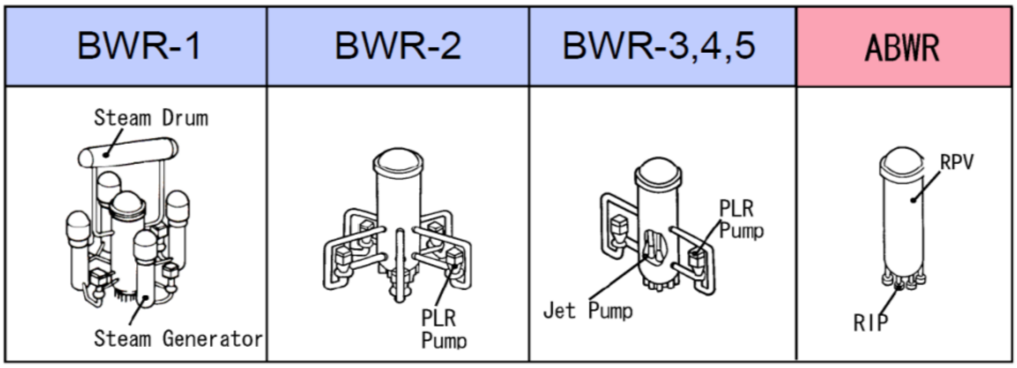


Figure 5.3.1.1-1: Evolution of Water Coolant Re-circulating Designs

The most recent evolutionary change, from the latest BWR to the standard ABWR, is the introduction of RIPs. The RIPs are located within the reactor and remove the requirement for re-circulation pipe work, valves and pumps located external to the reactor that are used in previous generations of BWR. The use of RIPs provides the following benefits:

- **Reduction in decommissioning waste.** The elimination of external pipes and valves contributes to a reduction in the volume of radioactive waste generated at decommissioning. Table 5.3.1.1-1 shows the weight of Reactor Recirculation System (RRS) main components in the BWR and ABWR. In the ABWR, most of the major pipe work with respect to RRS is installed in the PCV. As a result, the weight of RRS piping and valves has been reduced by approximately 40 tonnes and 20 tonnes, respectively. The weight of the RRS pumps has increased slightly in the ABWR compared to the BWR as the ABWR has ten RIPs compared to the BWR which has two Primary Loop Recirculation System (PLR) Pumps. However, overall, rationalising RRS from PLR Pumps in the BWR to RIPs in the ABWR has reduced the weight of pumps, piping and valves by 50 tonnes.

Table 5.3.1.1-1: Weight of RRS Main Components in BWR and ABWR Based on Hitachi-GE Experience

	Piping[tonnes]	Valve[tonnes]	Pump[tonnes]	Total[tonnes]
NPP A(BWR)	40	20	40	100
NPP B(ABWR)	0	0	50	50
Difference	-40	-20	10	**-50**

- **Reduction in the size of the PCV.** Removing components external to the reactor has reduced the space required and subsequently contributed to the size of the PCV being reduced (5.3.1.3 Evidence: Design of the Primary Containment Vessel).
- **Decrease in Operational Exposure to Workers**. The elimination of the external recirculation piping and valves means that there is no requirement for inspections during operation and lower radiation in work areas around the reactor vessel. The need to maintain an external recirculation system has also been removed which further reduces occupational exposures.
- **Improvement in safety.** The internal recirculation system means that an external pipe break with an associated loss of coolant is no longer possible. In the event of an accident the reactor core will remain covered in water. This extends the time available to initiate a response from other safety related systems installed to control abnormal reactor conditions.
- **Reduction in power consumption.** Rationalisation of piping, valves and jet pumps has reduced the amount of power required to drive the recirculation system.

5.3.1.2. Evidence: Evolution of the Primary Containment Vessel

The ABWR PCV incorporates a number of improvements over the BWR-5 design that will result in a reduction in the volume of radioactive waste requiring disposal. Table 5.3.1.2-1 provides details on how the functionality of the PCV has changed.

Table 5.3.1.2-1: Optimisation of PCV Primary Functions

	BWR-5		**ABWR**	
Function	**PCV**	**BSW**	**RCCV Liner**	**RCCV**
Withstand pressure	x			x
Leak prevention	x		x	
Shielding		x		x

The PCV of the BWR-5 utilises a free-standing SCV that delivers both the withstand pressure and leak prevention function. A separate BSW is provided within the R/B to meet the shielding requirements. The ABWR design utilises a RCCV with integrated internal RCCV liner; this configuration delivers all three functions [Ref-72].

The evolution of the PCV has resulted in the following benefits:

- a reduction in the mass of steel that will require disposal at decommissioning by approximately 40% (Figure 5.3.1.2-1);
- reduction in the R/B volume (Section 5.3.1.3); and
- by integrating the RCCV and RCCV liner the volume of concrete that requires disposal at decommissioning is reduced.

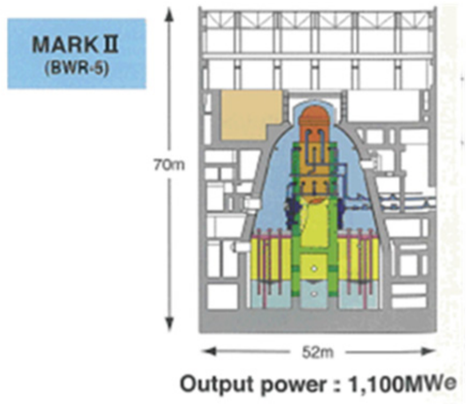
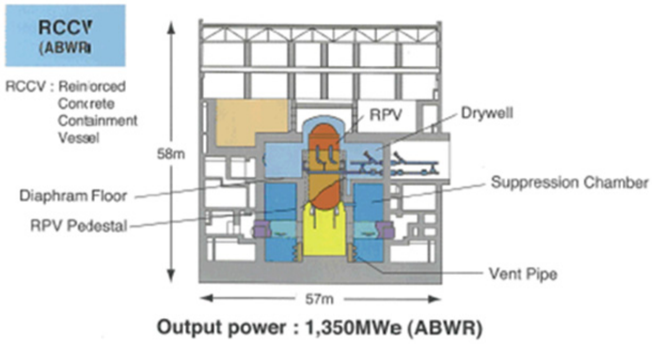
	
PCV (metal component): Approx. 3,850 tonnes	RCCV (metal component): Approx. 2,290 tonnes
Description: PCV main body, hatch, penetration portion, diaphragm floor, vent piping, γ-shield, RPV pedestal etc.	Description: RCCV liner, hatch, penetration portion, diaphragm floor, vent piping, γ-shield, RPV pedestal etc.

Figure 5.3.1.2-1: Comparison of BWR-5 PCV and ABWR RCCV

5.3.1.3. Evidence: Design of the Primary Containment Vessel

Figure 5.3.1.3-1 and Figure 5.3.1.3-2 illustrate the volume of the PCV in the BWR and the ABWR respectively. Compared with the volume of the BWR-5, the ABWR reduced the size of RCAs by approximately 15% as shown by Calculation 5.3.1.3-1 [Ref-34].

$$\text{Percentage volume reduction} = \left(1 - \frac{\text{Volume of PCV in ABWR}}{\text{Volume of PCV in BWR}}\right) \cdot 100$$

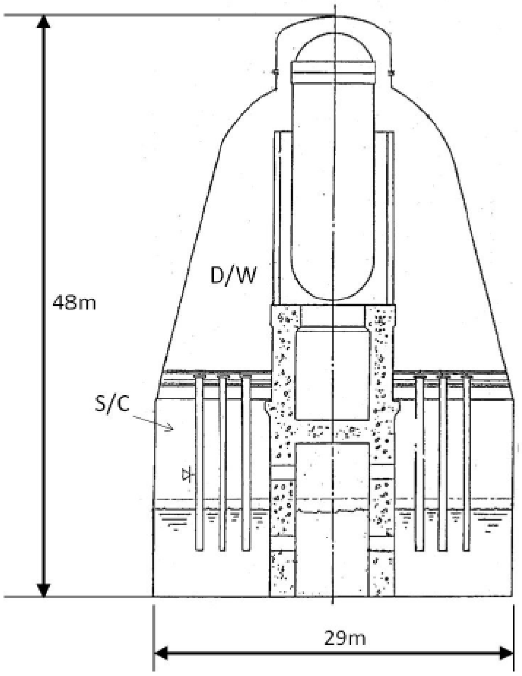
$$\text{Percentage volume reduction} = \left(1 - \frac{18{,}460\text{m}^3}{21{,}600\text{m}^3}\right) \cdot 100$$

$$\text{Percentage volume reduction} = 14.5\%$$

Calculation 5.3.1.3-1: Percentage Volume Reduction of PCV

The reduction in the size of the PCV has been achieved as a result of two improvements:

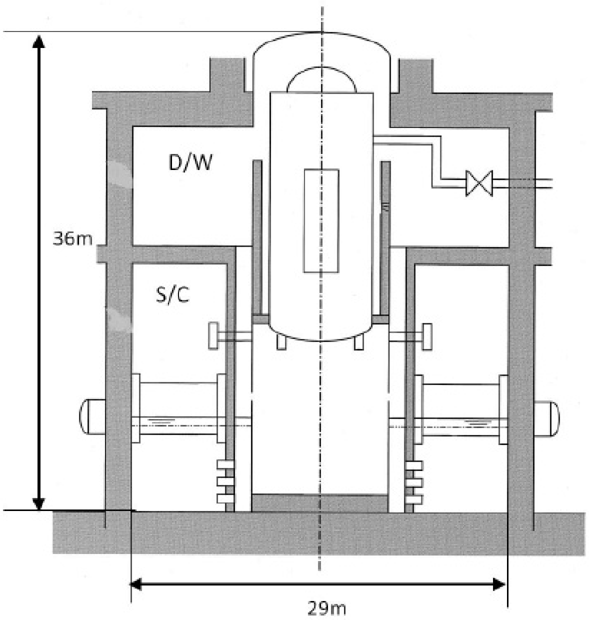
- **Introduction of RIPs**. As discussed in Section 5.3.1.1 eliminating the large bore pipe work associated with the coolant recirculation system allows the RPV to be located at a lower position within the PCV, and thereby enable the height of the PCV to be reduced.
- **Reinforced concrete containment vessel**. Replacement of the SCV used in the BWR with a RCCV in the standard ABWR. As shown in Figure 5.3.1.3-1 and Figure 5.3.1.3-2, the RCCV is cylindrical in shape, whilst the SCV has a circular cone structure. By adopting a cylindrical structure, the volume of the PCV has been reduced whilst the layout of equipment has been improved.



Main Volume of PCV in BWR-5

Area	Volume [m^3]
Dry Well (D/W) volume (including venting system)	10,300
Suppression Chamber (S/C) volume	11,300
Total volume	21,600

Figure 5.3.1.3-1: The Configuration of the PCV in BWR-5



Main Volume of PCV in ABWR

Area	Volume [m^3]
Dry Well (D/W) volume (including venting system)	8,600
Suppression Chamber (S/C) volume	9,860
Total volume	18,460

Figure 5.3.1.3-2: Configuration of the PCV in ABWR

5.3.1.4. Evidence: Techniques to Reduce Stress Corrosion Cracking

SCC results in the formation of cracks that if unidentified have the potential to result in sudden failure of an effected part. Mitigation of SCC focuses on material selection, improvements to the fabrication process (such as welding techniques) and the use of techniques to improve the operating environment, which is discussed in this section.

During normal operation of the reactor, water radiolysis results in the formation of oxygen and hydrogen peroxide. The presence of oxygen and hydrogen peroxide increases the ECP of the structural materials in the reactor which contributes to increased occurrences of SCC. Hitachi-GE has adopted hydrogen injection which results in a reduction in the ECP as a result of the recombination reaction shown below [Ref-35].

$$2H_2 + O_2 \rightarrow 2H_2O$$

$$H_2 + H_2O_2 \rightarrow 2H_2O$$

A consequence of injecting hydrogen into the feedwater is an increase in the amount of nitrogen-16 that is produced. Nitrogen-16 has a very short half-life and as such does not contribute to the gaseous radioactive discharges from the NPP into the environment. However, nitrogen-16 does contribute to worker dose and as such, Hitachi-GE has introduced NMCA which allows less hydrogen to be injected resulting in a reduction in concentrations of nitrogen-16 to levels equivalent to those achieved prior to the introduction of hydrogen injection. Further detail is provided in Section 5.1.6.1.

Monitoring of the ECP enables the performance of the combined hydrogen injection and nobel metal injection system to be optimised based on operational performance data.

Avoiding SCC reduces the requirement for maintenance and the replacement of reactor component parts. These activities would result in the generation of significant volumes of primary and secondary radioactive waste.

5.3.1.5. Evidence: Replacement of Pre-coated Filters

Filters are provided within the treatment systems for both condensate and LCW to remove iron crud (5.1 Claim 1). Early generations of BWR used pre-coat filters to perform this function. Pre-coat filters are coated in a resin that requires disposal once it is spent. A design change introduced the improvement of replacing pre-coat filters with HFF or pleated filters which are not coated with a resin [Ref-13]. The introduction of alternative filter types has resulted in the elimination of the spent powder resin waste stream. HFF and pleated filters also provide an increase in performance, improving the water quality. Design basis calculations have been undertaken to determine the volume of solid waste that has been avoided by the replacement of pre-coated filters (Table 5.3.1.5-1).

Table 5.3.1.5-1: Impact of Replacing Pre-coat Filters with HFF's for the CF

	Criteria		**Pre-coated filter**	**HFF**
Water quality	Fe crud concentration	Inlet	App. 8 to 10ppb	
		Outlet	Approx. 0.5 to 1.5ppb	≦Approx. 0.1ppb
Waste reduction	Spent resin volume		Approx. 30,000kg/Year (including crud)	0kg/Year
	Crud volume			Approx. 100kg/Year

5.3.1.6. Evidence: Review of Further Design Changes

Hitachi-GE has identified a number of design changes that also contribute to a reduction in the generation of radioactive waste [Ref-33]. A summary of these changes has been provided in Table 5.3.1.6-1.

Table 5.3.1.6-1: Summary of Design Changes that Contribute to Minimising Volume of Radioactive Waste

Items	Description
Introduction of larger capacity Safety Release Valve (SRV)	The number of SRVs was decreased from 18 to 16 valves by increasing the capacity of each SRV thus reducing the amount of maintenance and decommissioning waste.
Elimination of CUW motor operated valves	Removal of two motor-operated valves connected between the CUW and the feedwater lines.
Elimination of the third Main Steam Isolation Valve	The third MSIV was eliminated from the design by reducing the leak rate of MSIVs.
Elimination of pumps in the RHR system	The method for sealing RHR injection pipes with water was rationalised by using make up water condensate system (MUWC) pumps.
Elimination of S/P drain line for warm-up water	The main drain line for warm-up water is rationalised. This allowed for the elimination of the drain pipes connected to S/P.
Reduction of the number of valves monitored by the LDS	The valves to be monitored were reduced based on previous designs so that fewer valves are monitored.
Installation of pumps on points with lower temperature	Warm up pipes are eliminated by changing the composition of the CUW.
Adoption of boron racks in the fuel pool	Adoption of boron racks allowed for more efficient storage of SF which increased the capacity of the SF in the SFP. More efficient storage allowed for a reduction in the volume of the SFP area and thus the associated waste.
Elimination of interim loops in the R/B Cooling Water System (RCW)	Eliminate pipework that could become plated with activated products and subsequently require disposal as HAW.
Reduction in the capacity of Standby Gas Treatment System (SGTS) Fans	Reduce the capacity of SGTS Fan due to the low Secondary Containment leak rate and thus less plant requiring disposal at end of life.

It is also recognised that some of the design developments have resulted in the addition of plant items which may ultimately lead to greater volumes of waste being generated during maintenance and decommissioning. For example, moisture separator reheaters were introduced to improve the thermal efficiency of the plant but may become a waste during maintenance and decommissioning.

5.3.2.Argument 3b: Selection of Methods to Minimise Solid Waste Generation

The methods adopted by a future operator of the UK ABWR for operations and maintenance will influence the quantity of solid radioactive wastes requiring treatment, storage and disposal. The design of the UK ABWR includes a number of features that will allow any future operator to adopt an operating philosophy that will minimise the quantity of solid radioactive waste associated with routine operations and maintenance:

- Space is provided at key work locations within the designated areas to allow operators and maintainers to segregate wastes depending on their physical, chemical, radiological and biological properties. This will ensure that wastes do not become contaminated with substances that require more robust treatment, storage or disposal options (5.3.2.1 Evidence: Segregation of Waste).

- The provision of office accommodation outside of controlled areas which reduces occupancy of controlled areas and the associated generation of waste from office equipment and consumables (1.1.1.1 Evidence: Locate Offices Outside of Controlled Areas).
- The provision of space to store tools, scaffolding and maintenance equipment within designated areas to minimise the amount of equipment that is routinely taken in/out of designated areas for maintenance activities and subsequently disposed of (5.3.2.3 Evidence: Storage Facilities for Tools and Other Maintenance Equipment).
- The adoption of a flexible maintenance philosophy for non-critical items that allows items to be replaced on the basis of the function that they serve rather than on a pre-defined schedule (5.3.2.4 Evidence: Maintenance Philosophy).

The flexibility afforded by these features enables future operators to develop an approach that is appropriate to their operational needs and the regulatory requirements in force at the time. It is considered that these features support the demonstration of BAT for minimising solid radioactive waste generation at this time.

5.3.2.1. Evidence: Segregation of Waste

The Best Available Techniques for the Management of the Generation and Disposal of Radioactive Wastes report [Ref-32] recognises that effective waste collection, segregation, processing and storage is an effective means of reducing the activity, mass or volume of waste arisings.

The design of the UK ABWR enables a future operator sufficient flexibility to segregate, collect, store and process waste in a manner that allows BAT to be applied to the management and disposal of the waste. Methods for the collection, segregation, processing and storage of waste have been subject to detailed assessment which is reported in the BAT Optioneering report [Ref-43] and Radioactive Solid Waste Monitoring Requirements report [Ref-100] .

The Solid Waste Management System (SWMS) collects, processes and stores the wastes described in Table 5.3.2.1-1[Ref-36]:

Table 5.3.2.1-1: Summary of Solid Radioactive Waste and Spent Fuel Streams

No.	Title	Description	Category	Form	Arising during
1	Dry active waste	Miscellaneous dry, low activity wastes in various forms; including metals, concrete cloths, paper, etc.	Very Low Level Waste (VLLW)	Solid	Operations & Decommissioning
2	HVAC filters	Arising from filter changing in air treatment facilities from exhausts from R/B, T/B including high radiation), RW/B and S/B Buildings.	LLW	Solid	Operations & Decommissioning
3	Bead resin	Arising from the CD, LCW and HCW demineralisers; Styrene divinylbenzene copolymer matrix.	LLW (identified as borderline waste)	Wet	Operations & Decommissioning
4	Concentrates	Arising from the HCW/ Concentrated Waste System (CONW) evaporators comprises particulate and dissolved species.	LLW	Wet	Operations
5	Miscellaneous	Includes plastic sheets, paper, wood, cloth, oil and	LLW	Solid	Operations &

No.	Title	Description	Category	Form	Arising during
	combustible	activated carbon from Laundry Drain (LD) and Off-Gas systems.			Decommissioning
6	Miscellaneous non-combustible	Spent hollow fibre filters, metal, pipes, cables, lagging, gas filters, concrete and glass.	LLW	Solid	Operations & Decommissioning
7	Sludge (crud)	Arising from backwashing of various filters from the CF system and the LCW.	ILW	Wet	Operations
8	Powder resin	Arising from the CUW and FPC FD; cross linked polystyrene matrix. Contains particulate CP.	ILW	Wet	Operations & Decommissioning
	Ion exchange resin	Secondary waste arising from system decontamination of RPV, RIN and closed loop systems.	ILW	Wet	Decommissioning
9	Higher activity metals – control rods	Cruciform shape metallic construction containing stainless steel tubes in each wing of the cruciform filled with boron carbide powder. Hafnium is also employed to perform the same function of reactivity control.	HAW High Level Waste (HLW at arising, ILW at disposal)	Solid	Operations
10	Higher activity metals – fuel channels	Zircaloy box which surrounds the fuel bundle. Approx. 4.3m long and 15 × 15cm square. Channel boxes will stay with spent fuel so disposal “item” is a spent fuel assembly (fuel bundle in a channel box).	HAW (Remain with fuel elements)	Solid	Operations
11	Higher activity metals - others	Various reactor core components from SRNM and LPRM systems.	HAW (HLW at arising, ILW at disposal)	Solid	Operations
12	Contaminated and irradiated metal and concrete	Non-combustible, largely metal and concrete items. Some very large and requiring size reduction. Including SFP furniture (e.g. fuel racks).	LLW	Solid	Decommissioning
13	Contaminated and irradiated	Non-combustible, largely metal items. Some very large	ILW	Solid	Decommissioning

No.	Title	Description	Category	Form	Arising during
	metal	and requiring size reduction.			
14	Irradiated metal	Reactor core components and from areas subject to activation.	HAW (HLW at arising, ILW at disposal), ILW	Solid	Decommissioning
15	SF	Used fuel elements.	HLW	Solid	Operations

Wastes that have been segregated based on the waste forms and properties displayed in Table 5.3.2.1-1, are stored, transferred and processed independently of each other, to prevent mixing and cross contamination which allows them to be effectively treated based on their physical and chemical properties in accordance with the waste hierarchy.

Figure 5.3.2.1-1 and Figure 5.3.2.1-2 shows the waste flows of LLW and ILW respectively, and indicates how each waste stream is segregated based on waste form and properties and how this then allows for each waste stream to undergo the appropriate processing method before storage or disposal.

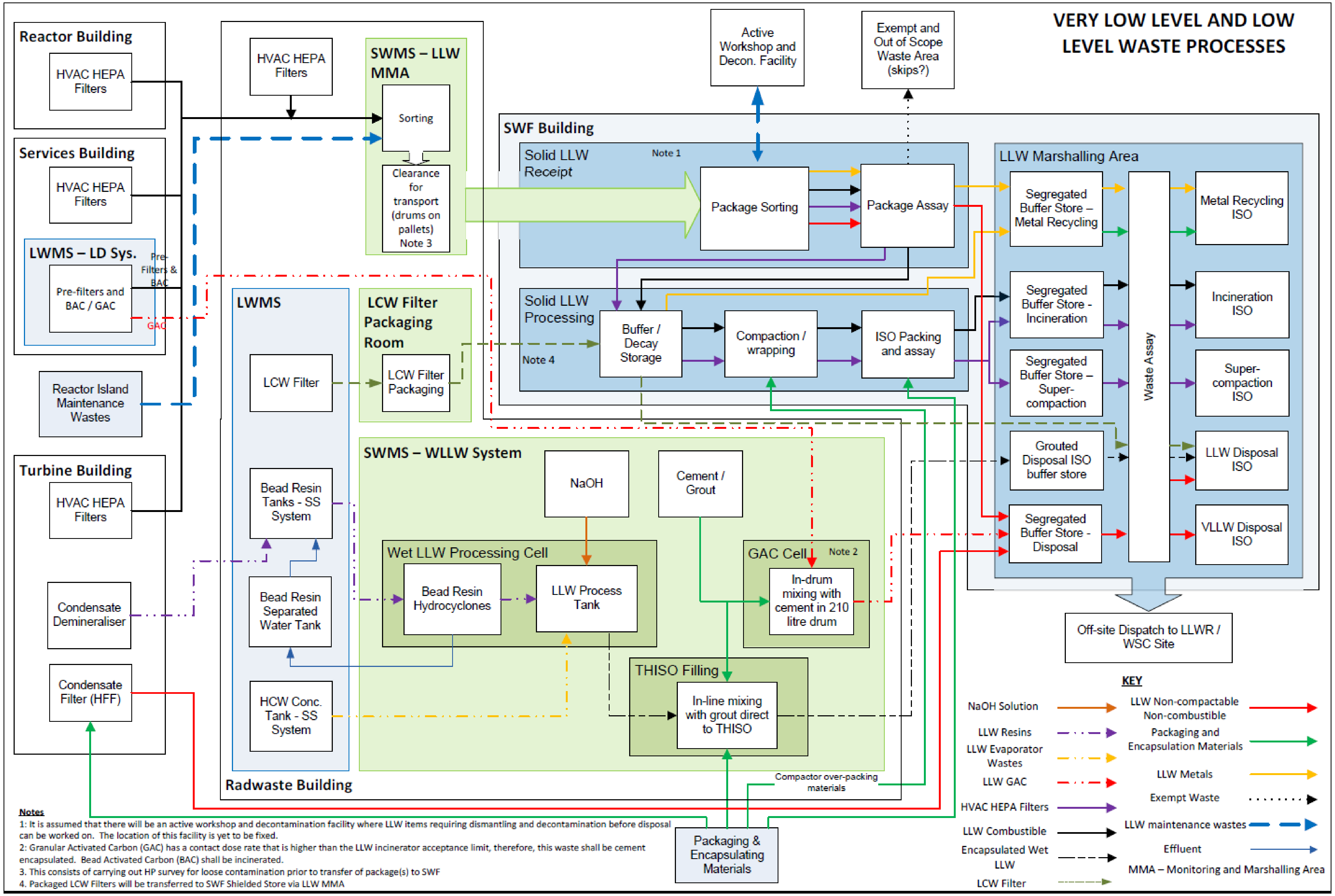


Figure 5.3.2.1-1: LLW Integrated Treatment System

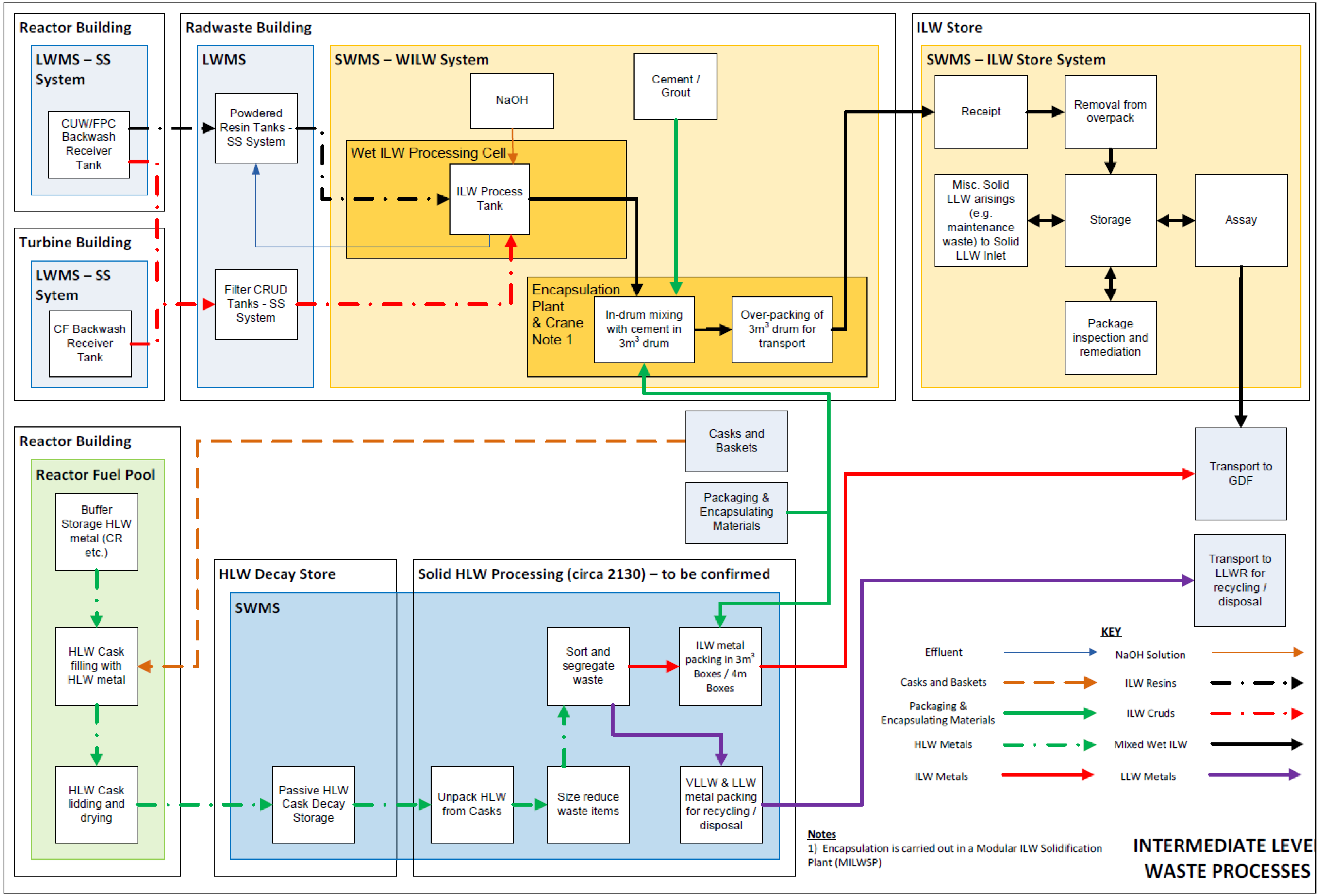


Figure 5.3.2.1-2: ILW Integrated Treatment System

5.3.2.2. Evidence: Locate Offices Outside of Controlled Areas

The UK ABWR design locates the majority of office accommodation outside of RCA. This minimises the number of operators routinely accessing RCAs and the volume of office related consumables routinely taken into RCAs.

Minimising the number of operator visits into controlled areas and reducing the volume of consumables that are taken into RCAs reduces the potential to generate secondary wastes. Management controls will be developed to ensure that the volume of consumables including PPE that needs to be taken into RCAs is minimised.

5.3.2.3. Evidence: Storage Facilities for Tools and Other Maintenance Equipment

Minimising the amount of maintenance equipment, tools, spares, consumables, protective clothing and packaging that are taken into RCA is recognised as RGP within the nuclear industry. The UK ABWR design therefore provides dedicated areas for the storage of maintenance equipment and tools. This will allow a future operator to minimise the amount of equipment and tools that require removal from RCAs following use and that have the potential to become radioactive waste. Retention of maintenance equipment and tools within RCAs after use will also obviate the requirement to bring new/replacement equipment and tools into the RCAs.

5.3.2.4. Evidence: Maintenance Philosophy

The maintenance philosophy for the UK ABWR ensures that non-critical items that have the potential to contribute to the generation of radioactive waste will only be replaced based on performance subject to recommendations from the manufacturer. Performance monitoring of HEPA filters is an example of how the UK ABWR has been designed to facilitate the implementation of this philosophy.

Replacement of HEPA filters. The UK ABWR design will allow a future operator to replace filters based on performance. Monitoring of the differential pressure across the filters will be provided to allow replacement of the filters at a pre-defined set point. This is expected to reduce the frequency of filter changes. In areas where high particulate matter loading is expected pre-filters will be used to minimise the potential to block HEPA filters extending the operational life of the HEPA filters.

Similar to HEPA filter replacement, the replacement of a FD element is determined by either a pressure drop higher than the specified value across the unit or by a conductivity rate at the FD outlet higher than the specified value during normal operation.

5.3.3. Argument 3c: Application of Volume Reduction Processes for Solid Waste

All solid radioactive waste is stored, transported and disposed of in containers that have been designed to meet the requirements of relevant legislation and regulatory guidance. In the majority of cases the waste is disposed of in the container in which it is transported to the waste management facility. Making the most efficient use of space in containers has the combined effect of reducing the size of storage facilities, decreasing the number of vehicle movements during transportation and minimising the demand on disposal capacity at appropriately permitted disposal sites (5.4 Claim 4: Selecting the Optimal Disposal Routes for Wastes Transferred to Other Premises).

The design of the UK ABWR includes dedicated facilities for the management, treatment and storage of solid radioactive waste. These facilities will include a provision for managing control rods that have reached the end of their useful life. The decision on how the control rods will be treated and subsequently stored will impact on the efficiency of storage and therefore the amount of space required. A study was conducted into the management of control rods which concluded that they would be held in casks for decay storage followed by size reduction prior to final disposal (5.3.3.1. Evidence: Size Reduction of Control Rods).

The UK ABWR will not include an on-site incinerator as the current best practice for volume reduction by combustion is to perform this task at an off-site industrial facility (5.3.3.2. Evidence: Incineration). The UK ABWRs solid radioactive waste management building has sufficient space and services to allow the introduction of a range of volume reduction techniques such as a waste compactor in addition, or as alternatives, to those specified in the design (5.3.3.3. Evidence: Solid Waste Compaction). This allows the future operator the flexibility to review the performance of techniques in the context of regulatory requirements and operating conditions prevailing at the time. The operator will be able to select the techniques most suited to current requirements. It is therefore considered that the combination of the defined volume reduction technology in the design and the flexibility to adopt alternatives in the future demonstrate the application of BAT at this stage.

5.3.3.1. Evidence: Size Reduction of Control Rods

Hitachi-GE has undertaken an assessment to determine how control rods will be managed including any requirement for size reduction once they reach the end of their useful life [Ref-75]. It was concluded that the preferred option for control rods would be decay stored in a cask system, or similar concept containers, for an extended period. Following decay storage, cutting would be carried out prior to disposal. Further development work and detail design would be required to fully establish and justify whether cutting of control rods in a wet or dry environment would be preferred. The UK ABWR design includes sufficient space to prevent the foreclosure of options for the management of control rods.

Control rods are a cruciform shape in cross-section which, if disposed of without any size reduction, have the potential to create a large amount of voidage in the selected disposal package, potentially resulting in a greater number of waste packages or larger sized waste packages. Use of size reduction techniques will ensure that the control rods are able to be packaged in a more efficient manner which ultimately reduces the volume of waste that will require disposal.

5.3.3.2. Evidence: Incineration

Incineration is a widely used and effective method of reducing the volume of combustible wastes. The majority of the radioactivity will either be captured in the ash which will be disposed of as solid waste or in abatement systems. The solid waste produced (ash and filters) will be significantly less than the original volume of waste consigned to the incinerator. Thermal treatments are reviewed in the Environment Agencies Requirements Working Group (EARWG) best practice guidance which concludes that thermal treatments are an effective volume reduction technique for combustible materials [Ref-37]. The EARWG best practice guidance states that volume reduction factors of up to 100:1 can be achieved.

In its Integrated Waste Management report, the Nuclear Decommissioning Authority (NDA) [Ref-38] has identified the opportunity to make more use of waste incineration which is currently used in the UK by existing nuclear facilities as a method of volume reduction. The main advantages of using incineration as a technique for treating combustible waste (once the waste is determined to be suitable for incineration) includes the significant volumetric reduction in the waste form. Volume reduction of waste provides the opportunity to improve the efficiency with which the Low Level Waste Repository (LLWR) is used, which is a national resource. Additionally, the option of incineration reduces the use of uncontaminated material for disposal (e.g. grout) and other resources that would have been used in the management, handling and long-term storage or disposal of the waste LLWR.

As with compaction, incineration which takes place off-site does not contribute to the minimisation of waste transferred to other premises but it does provide a means to significantly reduce the volume of waste ultimately requiring disposal. Incineration also provides a very effective route for the disposal of specific waste streams, such as oils, where the alternatives (e.g. encapsulation) result in the generation of large volumes of solid waste.

The Hitachi-GE Waste Treatment Study - BAT Optioneering report [Ref-43] assessed waste management routes including incineration for the management of waste streams. The assessment identified that incineration is the preferred management technique for the combustible waste stream. LLW filters were assessed according to the disposal costs per m^3 for three identified options:

- compaction;
- direct disposal to LLWR; and
- incineration.

Off-site incineration (with volume reduction techniques such as shredding and / or compaction as pre-transport treatment if this can be shown to have a cost benefit) was identified as the preferred option for LLW filters as it has the cheapest disposal costs per m^3.

Incineration was also assessed as an option for LLW resins. However, following the assessment, it was determined that bead resin would not meet the Waste Acceptance Criteria (WAC) requirement on total gamma/beta dose for incineration. For this reason, it was estimated that the incineration route for the LLW bead resin is not a viable option for this waste stream.

Should any aspect of the waste stream not meet the WAC for incineration, the contingency option is off-site compaction, as the volume of waste being disposed of would still be reduced. Direct disposal to the LLWR is the least favoured option for combustible wastes.

5.3.3.3. Evidence: Solid Waste Compaction

The EARWG identifies compaction as a widely used method to reduce the volume of dry radioactive waste. Compaction reduces the waste volume by reducing the amount of voidage in the waste [Ref-37].

Low force compaction can be carried out on-site and is a straightforward and effective volume reduction technology which is commonly used in the UK. Compaction does not produce any secondary solid wastes and any dust created can be contained by an appropriately designed secondary containment system. The typical volume reduction factor achieved using low force compaction is between 3 and 10 as stated in the EARWG best practice guidance.

High force compaction is typically carried out off-site at centralised facilities. High force compaction reduces the volume of waste to a higher degree compared to low force compaction. The EARWG Best Practice in Waste Minimisation report [Ref-37] states that it can achieve a volume reduction factor of up to 25. Although high force compaction is likely to take place off-site and does not contribute to the Claim that waste transferred to other sites will be minimised, it is recognised as the best practice for specified wastes and will reduce the volume that is ultimately disposed of.

Hitachi-GE has undertaken an assessment of waste treatment techniques [Ref-43]; this assessment recognised that it is industry best practice to implement low force compaction of LLW filters and LLW combustible and non-combustible soft waste as pre-transport treatment before incineration or final disposal.

Shredding and low force compaction are the preferred processing methods for soft combustible and non-combustible LLW. This process is considered a valid pre-treatment technique as it helps to reduce transport costs, minimises the environmental impact of transport, and incurs a lower cost for waste processing since this is determined by the volume shipped to LLWR.

5.3.4. Argument 3d: Solid Waste, Minimising the Quantity of Solidified High Chemical Impurities Waste

HCW contains impurities that increase the risk of corrosion and the associated generation of CPs. HCW must be treated to make it suitable for either reintroduction to the process water system or for disposal to the environment.

An evaporator is used to separate water from the impurities contained in the HCW. The evaporated water is collected, sampled and analysed prior to a decision on re-use or discharge to the environment. The concentrated liquor that remains in the evaporator contains all of the chemical impurities and the majority of the radioactivity associated with the HCW. This concentrated waste is solidified and subsequently disposed of as solid waste (5.3.4.1. Evidence: Solidification of Concentrated High Chemical impurities Waste). The concentrated liquid waste from the evaporator is solidified in cement.

Modifications to the drainage systems within the R/B has substantially reduced the quantity of HCW that is generated which has had a commensurate reduction in the quantity of concentrated liquor that is produced in the evaporator.

5.3.4.1. Evidence: Solidification of Concentrated High Chemical impurities Waste

Hitachi-GE has undertaken an assessment to demonstrate the application of BAT for the management of concentrated HCW liquors as part of its waste treatment study [Ref-43]. The assessment was based on the principle that concentrated liquid wastes will be co-managed with activated carbon and resin waste streams. This assessment has concluded that the preferred technique for the management of concentrated liquid wastes is solidification using an in-line cement immobilisation process.

The assessment considered the following options:

- solidification using cement immobilisation;
- polymer immobilisation; and
- off-site incineration.

Cement immobilisation was selected as the preferred option because:

- Drying of the resin followed by off-site incineration was the top ranking option at the workshop, subject to confirmation that the waste complied with the LLWR WAC. It was subsequently estimated that the bead resin would fail to meet the LLWR WAC requirement for limits on contact dose level, and for this reason it was concluded that this route is not a valid option for this waste stream. However, if alternative thermal treatment opportunities become available in the future then these should be considered.
- Polymer encapsulation was ranked second at the workshop by a small margin ahead of cement encapsulation. However, it was determined that polymer encapsulation has limitations around the maturity of the technique for LLW. There is also some uncertainty over the long-term stability of the waste form, and its acceptability against the LLWR WAC.

Cement immobilisation is a viable technique that is currently used in the UK. However, both the polymer immobilisation and off-site incineration options performed strongly during the Hitachi-GE assessment. A further assessment by a future operator is therefore required to demonstrate the application of BAT prior to the generation of resin wastes (FA8).

5.3.5. Argument 3e: Application of Decommissioning Techniques to Reduce the Activity and Volume of Decommissioning Waste

It is recognised that decommissioning operations will generate significant quantities of waste. A number of decommissioning techniques are available in order to reduce the activity and volumes of this waste. System decontamination using chemical decontamination techniques is carried out in order to remove radioactive contamination attached to inner surfaces of pipes. Decontamination after dismantling can lead to a lowering of the radioactivity category of the wastes, potentially reducing it to out of scope waste. The actual decommissioning of the UK ABWR and the selection of techniques will be the responsibility of the future operator however, at GDA it is considered to be adequate to demonstrate that a technique is available to undertake this task.

5.3.5.1. Evidence: System decontamination during decommissioning

System decontamination involves the removal of radioactivity, which is attached on the inner surfaces of pipes, with the use of chemicals. The chemical decontamination process is executed prior to dismantling of the plant and uses an oxidation-reduction reaction. Carrying out chemical decontamination prior to dismantling reduces the risk of the spread of contamination that could occur during cutting and disassembly of pipework. There are a number of techniques as displayed in Table 5.3.5.1-1.

Table 5.3.5.1-1: Chemical decontamination techniques

Classification of chemical decontamination method	Decontamination method	Removal target			DF	Secondary waste volume	Temperature of decontamination liquid
		Soft crud (iron oxide)	Hard crud (chromium oxide)	Base metal			
Oxidation-reduction dissolution decontamination method	CORD method*1	X	X		10-100	Very small	90°C
	HOP method*2	X	X		10-100	Very small	90°C
	T-OZON method*3	X	X		10-100	Very small	90°C
	DfD method*4	X	X	X	100	Small	90°C

*1 Chemical Oxidation Reduction Decontamination method (AREVA tech.)

*2 HOP method (Hitachi-GE Nuclear Energy tech.)

*3 Toshiba Ozone Oxidizing Decontamination for Nuclear Power Plants method (Toshiba tech.)

*4 Decontamination for decommissioning method (EPRI tech.)

5.3.5.2. Evidence: Decontamination after dismantling

After pipes and plant items are dismantled, decontamination using physical, mechanical and chemical methods can be applied in order to reduce the levels of contamination. Removing contamination can lower the waste category which aids with waste handling and disposal. A range of techniques are available including the following:

- Dry ice blasting;
- Ultrasonic cleaning;
- Water flushing;
- Scabbling;
- Shaving;
- Chemical fog;
- Chemical gel (TechXtract);
- Chemical foam;
- Strippable coatings; and
- Electropolishing.

Ultimately the selection of the decontamination techniques will be made by the future operator but should take the following into account:

- The material requiring treatment;
- Safety, ensuring there is no spread of contamination;
- Efficiency;
- Minimise work dose;
- Cost Effectiveness;
- Secondary waste generation; and
- Feasibility of Application.

5.4. Claim 4 – Selecting the Optimal Disposal Routes for Wastes Transferred to Other Premises

The design of the UK ABWRs radioactive waste building includes the space and services that are required to install the equipment necessary to undertake characterisation, treatment and storage of wastes to enable a future operator to select the optimal waste disposal route for radioactive solid wastes. For the purposes of GDA this report provides an indicative selection of disposal routes to demonstrate that the waste can be disposed of. The future site operator will determine the actual selection of disposal routes.

The Arguments presented in support of this Claim demonstrate compliance with the standard BAT conditions [Ref-2]:

- Condition 2.3.3 (b) 'Characterise, sort, segregate solid and non-aqueous liquid wastes, to facilitate the disposal by optimised disposal routes.'
- Condition 3.1.3. 'The operator shall dispose of each form of solid and non-aqueous liquid radioactive waste by an optimised disposal route for that waste form.'

This is also considered to fulfil the following requirement of the P&ID [Ref-1]:

- Selecting optimal disposal routes (taking account of the waste hierarchy and the proximity principle) for those wastes.

The UK ABWR design contains a range of features that contribute to the substantiation of this Claim including:

- An agreement in principle will be obtained for the disposal of lower activity wastes that will be generated during the lifetime of the UK ABWR.
- Disposability assessments will be prepared for HAWs.

In developing the Arguments presented to demonstrate the validity of Claim 4 the REPs have been taken into account. The following REP is considered to be specifically relevant to this Claim:

- **Principle RSMDP7** 'When making decisions about the management of radioactive substances, the best available techniques should be used to ensure that the resulting environmental risk and impact are minimised.'

5.4.1.Argument 4a: Provision of Waste Management Facilities

A range of facilities and equipment are required to ensure the effective and efficient use of available waste management routes. The design of the UK ABWRs radioactive waste building includes the space and services to install the equipment necessary to undertake the characterisation (5.4.1.1 Evidence: Waste Characterisation and Assessment Facilities), sorting (5.4.1.2 Evidence: Segregation and Sorting Facilities), treatment (5.4.1.3 Evidence: Waste Treatment Facilities) and storage (5.4.1.4 Evidence: Waste Storage Capacity) of waste prior to consignment to an appropriately permitted waste management service supplier. These facilities reflect the outputs of the waste strategy that has been developed for the wastes that will be produced by the UK ABWR. The specification and selection of equipment used within the Rw/B will be undertaken by the future operator (FA6). The current size and configuration of the Rw/B is considered to offer future operators the flexibility to select and implement techniques that will reflect its operational needs and the regulatory requirements in force at the time [Ref-43]. The provision of such a flexible facility is considered to represent BAT at this stage.

5.4.1.1.Evidence: Waste Characterisation and Assessment Facilities

The design of the UK ABWR will include sufficient space to undertake sampling and non-destructive assay of solid waste prior to transfer off site for disposal or long term storage [Ref-43]. As well as waste assay to determine the radiological properties of the waste, the characterisation and assessment facilities enable a future operator to determine the physical and chemical properties of the waste. This allows for the effective segregation of waste and the identification of suitable waste routes based on the characteristics of each waste group. Waste management practices proposed for the UK ABWR will produce packages consistent with those already produced in the UK nuclear industry. Techniques for sampling and non-destructive assay currently available in the UK are therefore considered to be suitable to be used for the characterisation of solid wastes from the UK ABWR. The provision of sufficient space for waste characterisation within the design of the UK ABWR provides flexibility for a future operator to select the specific technique through the application of BAT. This will also provide a future operator the opportunity to consider the most up to date techniques as part of any assessment (FA8).

5.4.1.2. Evidence: Segregation and Sorting Facilities

The design of the UK ABWR provides sufficient space to allow a future operator to carry out those techniques detailed in the Radioactive Waste Management Arrangements [Ref-36] [Ref-43]. The system allows for the segregation and sorting of waste into diversified waste streams prior to the appropriate treatment and/or disposal option for each waste stream as detailed in Section 5.4.1.3. (Evidence: Waste Treatment Facilities) and Section 5.4.2. (Argument 4b: Optimal Disposal Route Selection). Details relating to the selection of techniques that will be adopted for the segregation of wastes are provided in Section 5.3.2 (Argument 3b: Selection of Methods to Minimise Solid Waste Generation).

5.4.1.3. Evidence: Waste Treatment Facilities

The techniques required for the treatment of waste have been selected for GDA (5.4.2.1 Evidence: Waste Treatment Techniques and Disposal Routes) to ensure effective and efficient treatment of waste and to enable waste to be disposed of in accordance with the Radioactive Waste Management Arrangements [Ref-36].

Facilities are required within the design of the UK ABWR to receive, process, package/condition and export the following waste types:

- Wet ILW (spent resin and crud wastes);
- Solid ILW (activated metal waste);
- Wet LLW (spent resin, sludge, activated carbon and concentrated liquid wastes); and
- Solid LLW (combustible, non-combustible wastes and HVAC filters).

Following optioneering, it has been determined that there will be a Rw/B for solid waste and a combined wet ILW and LLW treatment facility [Ref-98]. It is deemed that these facilities ensure that sufficient space is available for waste management. However, a future operator has the option of combining facilities to service a number of units which may provide some advantages in terms of minimising the duplication of common processes and the efficiency of operations.

A summary of how the waste management facilities will interface with other parts of the UK ABWR is provided in Figure 5.4.1.3-1.

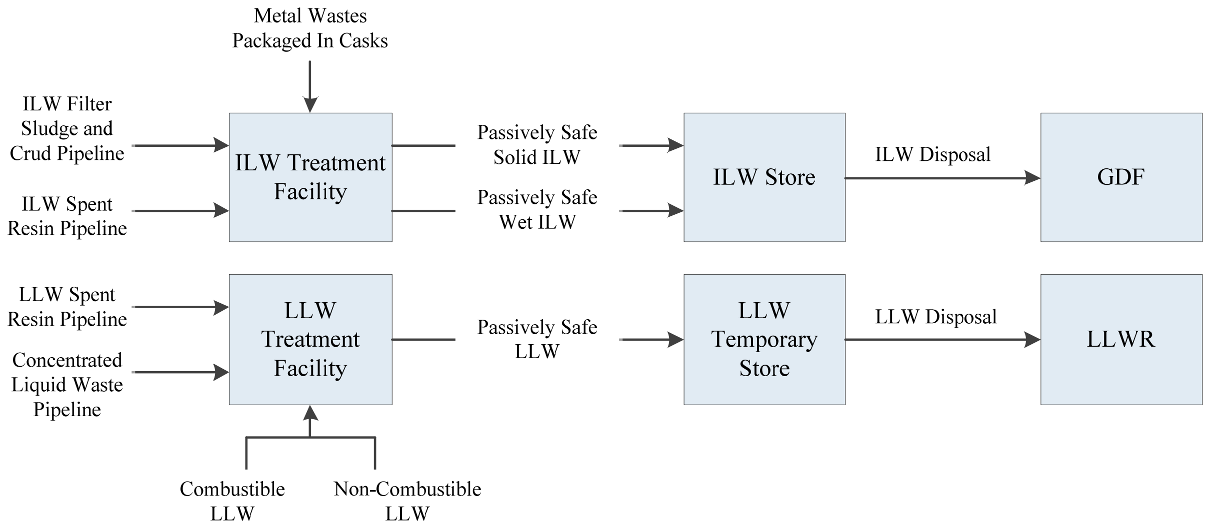


Figure 5.4.1.3-1: Summary of the Flow of Waste into and Out of the Waste Treatment Facilities

Solid Waste Facility (SWF)

The solid waste facility (SWF) will process the solid LLW into a form that will allow incineration or disposal in line with the selected disposal routes detailed in Section 5.4.2. (Argument 4b: Optimal Disposal Route Selection). The techniques have been selected to ensure that waste packages will be compliant with the WAC proposed as part of the Waste Service Contract (WSC) for LLWR.

The building is deemed to allow sufficient space to accommodate the necessary plant items that are required for dry-solid LLW receipt, sorting, treatment and packaging. Further detailed design considerations are included in the system design description [Ref-78] and the PCSR Chapter 18 [Ref-58].

The main plant items in the solid waste facility (SWF) include:

- sorting conveyor array;
- combustible waste shredder; and
- low force compactor.

ILW Store

The ILW storage facility will decay store solidified wet ILW in 3m^3 drums for an extended period prior to disposal to the GDF in accordance with Radioactive Waste Management Limited (RWM) requirements. Solid ILW will be stored in 3m^3 boxes. Further detailed design considerations are included in the System Design Description [Ref-79].

Wet –solid LLW and ILW Processing System

The wet-solid LLW and ILW processing system will treat both wet-solid LLW and ILW separately dependent on their treatment technique. The wet-solid LLW treatment part of the facility will process spent resins and concentrated liquid waste from the evaporator to allow disposal. For the purposes of the GDA, space requirements are based on the requirement to immobilise wet-solid LLW using a cementation process. However, it is recognised that a future operator may adopt an alternative treatment process such as a polymer based immobilisation technique (FA8).

The wet-solid ILW treatment part of the facility will process the wet-solid ILW into an immobilised form compliant with NDA RWM requirements using a cementation technique. For the purposes of the GDA, space requirements are based on the requirement to immobilise wet-solid ILW using a cementation process. However, it is recognised that a future operator may adopt an alternative treatment process such as a polymer based immobilisation technique. The space provided within the design of the UK ABWR is considered to be sufficient to allow a future operator the flexibility to adopt an alternative treatment technique if this is demonstrated to be BAT during subsequent assessments.

The facility allows sufficient space to accommodate the necessary plant items that are required for wet-solid LLW and wet-solid ILW receipt, preparation for treatment, immobilisation and disposal (for wet-solid LLW) or transfer to the ILW store (for wet-solid ILW). Further detailed design considerations are included in the System Design Description [Ref-76][Ref-77] and in PCSR Chapter 18 [Ref-58].

The main plant items in the wet-solid LLW treatment part of the facility include:

- a LLW process tank;
- cement grout preparation equipment (powder feeder, grout feed hopper and grout pump); and
- in-Line Mixer.

The main plant items in the wet-solid ILW treatment part of the facility include:

- ILW resin batch tanks;
- ILW crud batch tank;
- Pre-blended cement grout preparation equipment (powder feeder, mixer, grout feed hopper and grout pump);
- Pre-blended cement powder feed system; and
- Modular ILW solidification plant.

5.4.1.4. Evidence: Waste Storage Capacity

The design of the UK ABWR includes the capacity to temporarily store waste as detailed in the Radioactive Waste Management Arrangements [Ref-36]. The capacity of the storage facilities is sufficient to allow a future operator to optimise storage times through the application of BAT.

LLW

LLW is consigned for transfer off-site in batches. To accrue the required number of packages that comprise a batch, the design of the UK ABWR includes an on-site buffer store. During normal operations the storage period shall be determined based on the batch size required to allow efficient disposal to the waste receiver. However, the size of the store has been determined based on the volume of waste that will be generated in a two year period. This additional capacity will provide sufficient flexibility for a future operator to optimise the management of waste consignments and to allow for any disruption to the transport system or problems at the disposal facility which could temporarily restrict or curtail the acceptance of waste consignments.

ILW

Hitachi-GE is undertaking an assessment to determine the most appropriate method for storing ILW on the generic site for the purpose of GDA. The assessment will assume that ILW will be stored on-site in facilities, whose characteristics will be suited to the specific packages/containers which are stored in them. As a minimum requirement, the stores will be built to UK codes and standards and will represent RGP for the storage of this type of waste.

SF

The store arrangement for SF will depend upon the option chosen for its management. There are several options available for long term on-site storage of SF which are described in the Radioactive Waste Management Arrangements [Ref-36]. Further detail is also provided in PCSR Chapter 32 [Ref-89]. The option will be selected by a future operator and will be determined through the application of BAT (FA11).

5.4.2. Argument 4b: Optimal Disposal Route Selection

The UK Government and Devolved Administrations published a revised policy for the long term management of solid LLW in 2007. This policy recognised that the previous preference for disposing of LLW from nuclear sites to the national LLWR was no longer sustainable and that alternatives for the management of these wastes were required. The revised policy requires nuclear operators to consider a range of options when developing plans for the management of solid LLW. These options are to be based on the waste hierarchy and are to take into account a broad range of environmental and sustainability principles in addition to those related to the risk of exposure to potentially harmful ionising radiation.

Since 2007 the nuclear industry and its suppliers have made significant progress in developing alternatives to the disposal of LLW to the national LLWR. A range of techniques have been implemented that allow LLW to be:

- minimised at source;
- re-used/recycled;
- volume reduced prior to disposal; and
- disposed of at alternative sites to the national LLWR.

The design of the UK ABWR, and the Radioactive Waste Management Arrangements [Ref-36] developed to manage the waste, recognise that a range of waste management options are available for the management of LLW that will arise during the operation of the NPP. Strategic consideration of options related to the provision of on-site waste treatment facilities has concluded that incinerators, metal treatment and disposal facilities will not form part of the generic design for the UK ABWR. This decision is consistent with the findings of a number of strategic studies that are discussed further in Section 5.4.5 (Argument 4e: Compatibility of Existing UK Waste BAT Studies). Evidence is provided to support the use of selected waste management techniques (including those provided by LLWR) for the disposal of LLW (5.4.2.1 Evidence: Waste Treatment Techniques and Disposal Routes). It is recognised that this is a high level assessment that will require further assessment by any future operator (FA7).

The decision to select specific waste management options also takes account of recent permitting programmes at nuclear sites and recognises that there is now an established market for these treatment and disposal techniques. The compatibility of the UK ABWRs waste with the WAC of commercially available disposal routes is discussed further in Argument (5.4.3: Argument 4c: Agreement in Principle for Waste Routes - Lower Activity Wastes).

The design of the UK ABWR solid waste management facilities allows future operators a high degree of flexibility in the selection and deployment of LLW treatment techniques as discussed in Section 5.4.1 (Argument 4a: Provision of Waste Management Facilities). This will enable them to review the availability and performance of LLW treatment/disposal techniques that are available at the time and to implement any measures required to ensure that their LLW is compatible with the WAC for such techniques (FA6). The outcomes of such reviews will be included in the waste management plans and will demonstrate that BAT is being applied to the management of these wastes.

5.4.2.1. Evidence: Waste Treatment Techniques and Disposal Routes

The techniques required to manage waste from the UK ABWR have been assessed in the Hitachi-GE waste treatment BAT assessment [Ref-43]. A summary of the conclusions from the assessment has been provided below:

Solid LLW

The preferred waste management techniques for solid LLW are:

- Initial size reduction of filters (for example low force compaction), followed by transfer to an off-site facility either for supercompaction or incineration, dependent on the filter design.
- Combustible waste will be subject to low force compaction before being transferred off-site for incineration at an appropriately permitted site.

- Recyclable metals will be transferred off-site for recycling at an appropriately permitted metal melting facility.
- Non-combustible and non-compactable waste (including metals unsuitable for recycling and grouted waste) will be sentenced for disposal at an appropriately permitted site.

Solid ILW

The preferred waste management techniques for the solid ILW streams are:

- Activated metals will be dealt with on a case-by-case basis. Where appropriate and worthwhile, the metals will initially be stored unpackaged in the SFP to allow for decay storage to reduce the radioactivity. Decontamination would have no effect because the waste is activated and the radioactivity is integral to the waste, not on the surface. The metals will then be retrieved, size reduced (if necessary) and packaged for final disposal in an appropriate container and stored onsite in the storage facility, pending final disposal to the GDF.
- Activated solid radioactive waste that will arise from decommissioning will be dealt with in the same manner as activated metals (see above) although it will be stored in situ rather than in the SFP.
- Contaminated solid waste that will arise from decommissioning will undergo chemical decontamination in order to prevent the spread of contamination. After which, the waste will be size reduced where necessary and packaged into an appropriate disposal container. The waste will be stored in the storage facility, pending final disposal to the GDF.
- The storage facility will comprise of a segregated area designated for decay storage until sufficient decay has taken place to enable processing, transport, and storage.

Wet LLW

The preferred technique for the management of wet LLW (resins, concentrated liquid wastes, and activated carbon) is cement immobilisation, using in-line mixing of waste and cement. The preliminary assessment identified that two other options may perform better if uncertainties identified during the assessment could be resolved [Ref-43]. These options were drying followed by incineration and polymer immobilisation. Subject to these techniques being further developed the assessment recommended that further assessment should be undertaken by a future operator prior to implementation of the cementation technique (FA6).

Wet ILW

The preferred technique for the management of wet ILW (resins and crud) is cement immobilisation and storage / disposal. Cement immobilisation is an established technique in the UK nuclear industry. The assessment did identify vacuum drying as a high performing option although further development would be required to demonstrate its viability compared to cementation. A future operator would therefore be required to carry out a further assessment prior to implementation of cement immobilisation to determine the application of BAT for wet ILW (FA6).

The techniques identified above demonstrate that viable disposal routes are available for the different UK ABWR waste forms. For the purposes of GDA the viability of the selected waste management technique for VLLW and LLW has been demonstrated using the services provided by LLWR through an agreement-in-principle (5.4.3 Argument 4c: Agreement in Principle for Waste Routes). The service from LLWR includes four main options, as detailed in the Radioactive Waste Management Arrangements document [Ref-36]:

- Metallic waste processing service;
- Combustible waste processing service;
- Compaction waste processing service; and
- LLW disposal service.

The range of disposal routes identified allow for disposal options other than disposal to the national LLWR which is consistent with the 2007 UK policy for the long term management of solid LLW [Ref-95].

However, a future site operator will need to determine the specific waste management route through the application of BAT. Site specific considerations will need to be considered when selecting the optimal waste management option such as the proximity principle.

5.4.3.Argument 4c: Agreement in Principle for Waste Routes - Lower Activity Wastes

Each of the routes for solid and non-aqueous lower activity radioactive wastes has a series of requirements that the consignor of the waste must fulfil before it can be accepted. Compliance with the WAC is a requirement of the terms and conditions agreed between contracting parties. Compliance with WAC is also a requirement of the Environment Agency's standard environmental permit template for disposals of radioactive waste from nuclear licensed sites as they are 'instructions' given by the person to whom the waste is consigned. Hitachi-GE, as the requesting party for GDA will not dispose of waste. However, the GDA process requires that the requesting party obtains an 'agreement-in-principle' to dispose of waste via each of the waste routes. This process provides assurance that the waste generated from operating, maintaining and decommissioning of the UK ABWR will comply with the WAC for the waste routes and can be disposed of by any future operator.

Hitachi-GE has been engaging with the suppliers of waste management services for solid and non-aqueous radioactive waste in the UK and has obtained an agreement in principle (5.4.3.1 Evidence: Agreement in Principle) for the following waste routes:

- metallic waste for physical decontamination and recycling;
- combustible waste for volume reduction by incineration;
- VLLW for disposal at appropriately permitted commercial landfills;
- compaction of compressible lower activity waste followed by disposal in the national LLWR; and
- disposal of non-compressible lower activity waste in the national LLWR.

The agreements in principle will demonstrate that lower activity wastes that will be produced by the UK ABWR are compatible with the range of waste routes and services available in the UK for such wastes. This is considered to represent BAT for the GDA process. The final choice of waste route and the quantity of waste to be consigned will be determined by the future operator (FA8).

5.4.3.1.Evidence: Agreement in Principle

For the purposes of GDA, Hitachi-GE has made the assumption that LLWR will provide all waste services through a WSC. Through the WSC, LLWR provides a range of waste management options to ensure lower activity wastes are managed in the most efficient manner possible with the ultimate objective of preserving the disposal capacity at the LLWR. To facilitate this objective Hitachi-GE will ensure that the requirements of the waste hierarchy are applied to the management of these wastes. Hitachi-GE has submitted a waste enquiry form to LLWR and has established an agreement in principle with the waste disposal routes identified in Section 5.4.2. (Argument 4b: Optimal Disposal Route Selection). A letter from the LLWR giving ‘Agreement in Principle’ for the VLLW and LLW has been received by Hitachi-GE [Ref-68] [Ref-69]. Hitachi-GE has re-assessed the waste specific activities as presented in this document following an update of the generic site source term in relation to the relevant LLWR WAC and has determined that they remain within the acceptance criteria [Ref-70].

The LLWR range of waste services has been developed by the National Waste Programme team to support implementation of the 2007 Strategy [Ref-95]. For the purposes of the GDA submission, Hitachi-GE will use the services offered by LLWR, although, any future site operators may choose their own arrangements based on the site specific requirements and the demonstration that they have selected the optimal disposal route at the time. The future operator will carry out a detailed assessment of a range of waste routes, not just those supplied through LLWR, and will demonstrate BAT and the application of the waste hierarchy in the final selection of each waste route (FA8). A site operator will need to make business decisions, which will also include commercial and logistical requirements which may indicate that there are other ways of dealing with specific wastes. For example, the site operator may decide to contract directly with a waste service supplier rather than utilising the LLWR contract arrangements.

5.4.4. Argument 4d: Disposability Assessments for Higher Activity Wastes

Some of the solid radioactive wastes that will be created in the UK ABWR will be too radioactive to be disposed of via existing routes; HAW and SF will be stored until a long-term management solution is available. Any treatment and storage arrangements must accord with the current management and/or disposal concepts for these wastes. RWM is the source of authoritative guidance regarding management and disposal concepts. RWM undertakes assessments to determine the degree to which proposals for the management of HAW and SF accords with management and disposal concepts.

Hitachi-GE has engaged extensively with RWM during the development of the GDA submission for the UK ABWR. Disposability assessments have been prepared for the following:

- Disposal of SF in the proposed national GDF (5.4.4.1 Evidence: Disposability Assessment – Spent Nuclear Fuel).
- Disposal of HAW in the proposed national GDF (5.4.4.2 Evidence: Disposability Assessment – Intermediate Level Waste).

The disposability assessments demonstrate that HAW that will be produced by the UK ABWR will be compatible with the range of waste routes and services available in the UK for such wastes. For the SF the disposability assessment demonstrates that there are methods currently available which are used successfully in other countries and could be designed and implemented to comply with UK regulatory and disposability requirements. This is considered to represent BAT for the GDA process. The final choice of waste and SF routes and the quantity of waste to be consigned will be determined by the future operator (FA8).

5.4.4.1. Evidence: Disposability Assessment – Spent Fuel

Hitachi-GE is required to obtain a view from RWM as to whether the HAW can be disposed of in line with the plans for a GDF in the UK. As part of obtaining that view, Hitachi-GE has provided RWM with a suite of information in support of Disposability Assessment Submission for ILW and SF [Ref-83] and [Ref-84] to allow them to undertake their assessments.

The Radioactive Waste Management Arrangements document [Ref-36] describes the characteristics of SF from a UK ABWR and the options for conditioning, storage and disposal. These options are based upon good practice as currently applied in other countries (including one option planned for implementation at Sizewell B in the UK) and which would be considered as BAT at this time. High level optioneering on SF storage [Ref-87] concluded that the dry cask storage system should be adopted for GDA. However, the selection of the appropriate option by the future operator, when developing its plans for conditioning, storage and disposal will be subject to the application of BAT within the decision making process (FA11).

At this stage several options for packaging the SF for storage and eventual disposal will be included because the preferred method has not yet been determined. The final choice will involve many other aspects related to specific site requirements, including storage arrangements, the need for long term cooling prior to disposal, etc. Options for the on-going management of the SF include:

- Dry cask;
- Multi-purpose container;
- Modular vault dry storage system;
- KBS-3 container; and
- On-site fuel pool option.

5.4.4.2. Evidence: Disposability Assessment – Intermediate Level Waste

In order to demonstrate that the waste produced by the UK ABWR can be disposed of and managed in the UK regulatory system a single option has been selected to enable more detailed studies and analysis. At this stage the option of cement encapsulation (for solid items) and solidification (for wet/slurry wastes) into unshielded stainless steel containers has been selected as the packaging method to be adopted for a disposability assessment by RWM.

The ILW will be dealt with in according with the requirements of RWM and has been subject to disposability assessments during the GDA process. For later, site specific waste streams and packaging methods a Letter of Compliance (LoC) process is used to obtain endorsement. Assessments to support this process are currently underway. The adoption of the noted solution for GDA will not preclude a future operator from selecting another method, subject to completion of relevant BAT assessments, business case analysis and LoC assessments.

5.4.5. Argument 4e: Compatibility of Existing UK Waste BAT Studies

Selection of appropriate waste routes is an important element of demonstrating that waste management practices form part of an integrated strategy that is focussed on waste minimisation, the application of the waste hierarchy and demonstrating the application of BAT. A series' of strategic BAT assessments have been prepared by LLWR at the behest of the NDA to examine the degree to which certain waste routes underpin the development of integrated waste strategies for producers of radioactive waste and support delivery of the waste hierarchy. These assessments adopted a systematic, robust and transparent approach to determining the strategic BAT option for groups of radioactive waste with broadly similar characteristics.

Hitachi-GE has undertaken a series of assessments to determine the degree to which the findings of these studies are applicable to the following types of waste that will be generated by the UK ABWR:

- Metal waste that has radioactive contamination on its surfaces (5.4.5.1: Evidence: Review of LLWR Metallic Waste Strategic BAT Assessment).
- Combustible wastes that are lightly contaminated with beta and gamma emitting radionuclides and/or very lightly contaminated with alpha emitting radionuclides (5.4.5.2: Evidence: Review of LLWR Combustible Waste Strategic BAT Assessment).
- Waste with very low levels of radioactivity (5.4.5.3: Evidence: Review of LLWR VLLW Strategic BAT Assessment).

It is concluded that the LLWR assessments are applicable to the UK ABWR and that BAT is demonstrated at a strategic level for metallic wastes, combustible wastes and wastes with very low levels of radioactivity.

5.4.5.1. Evidence: Review of LLWR Metallic Waste Strategic BAT Assessment

The LLWR strategic BAT assessment for the management of metal waste [Ref-39] evaluated options for the management of metallic LLW against a number of safety, environmental, economic, and technical attributes. The assessment concluded that in the short term the best option is waste processing in an overseas metal recycling facility and in the longer term the development and use of UK metal recycling facilities. The report identified a number of advantages associated with this option including:

- Secondary solid waste generation is significantly reduced compared to the on-site or national disposal options.
- The option allows a significant quantity of material to be recycled thereby reducing the overall demand on raw materials and energy requirement of processing raw materials.

Hitachi-GE has identified in its waste treatment assessment [Ref-43] that recyclable metals will be consigned under the LLWR WSC for recycling. Whilst, for the purposes of GDA, Hitachi-GE does not explore the ultimate route for treatment as this will be conducted by the future operator (FA7); the conclusions of the strategic BAT assessment are deemed to be consistent with the strategy Hitachi-GE will adopt for metallic wastes. That is to say that metallic wastes that comply with the WAC for metal treatment waste management facilities shall be sent off-site for treatment with the aim of recycling as much of the waste stream as possible.

5.4.5.2. Evidence: Review of LLWR Combustible Waste Strategic BAT Assessment

The LLWR strategic BAT assessment for the management of combustible LLW [Ref-40] evaluated options against a number of safety, environmental, economic, and technical attributes. The overall recommendation was to identify thermal treatment contractors with a view to establishing contractual and commercial arrangements for the disposal of this waste form at the earliest opportunity. Additionally it was recommended that the practice of compaction be continued for those waste streams that could not be readily routed for thermal treatment.

The conclusions of the strategic BAT assessment are deemed to be consistent with the strategy Hitachi-GE has developed for the management of combustible wastes [Ref-54]. For the purposes of GDA it is assumed that combustibles will be sent, under the LLWR WSC for combustible materials, for incineration. The future operator will have the flexibility to determine the incineration facility selected based on site specific factors and through the application of BAT.

5.4.5.3. Evidence: Review of LLWR VLLW Strategic BAT Assessment

The LLWR strategic BAT assessment evaluated the relative merits of options for the long-term management of VLLW [Ref-41]. The main conclusions of this report were:

- The application of the waste hierarchy to VLLW is supported and reinforced through the study.
- Whilst the performance of the disposal options are broadly comparable:
    - Options that represent existing co-disposal facilities generally scored well together with new disposal facilities located on-site or adjacent to nuclear licensed sites.
    - On-site disposal scored better than other new disposal facility options.
    - There is little differentiation between new local, regional and national facilities.
    - Use of LLWR at Drigg scores consistently low.
- Depending on the waste stream, site specific issues and the WAC of any of the options, VLLW may require disposal via an option that does not score highly in this assessment. This may still be consistent with the strategic BAT assessment where specific arguments and justifications can be made.

The conclusions made by the LLWR report are deemed to be consistent with the strategy Hitachi-GE will adopt for VLLW.

Dry active waste is the only VLLW stream identified for the UK ABWR. This is a mixed waste that will arise during reactor operations and decommissioning. The waste consists of contaminated PPE, monitoring swabs, plastic, equipment, structures and contaminated plant. These wastes will require specific removal, handling, sorting and size reduction techniques depending on their physical form and characteristics prior to treatment. The preferred strategy for this mixed dry active VLLW is to recycle the metallic portion where practicable and to dispose of the remainder, following pre-compaction if possible, to permitted disposal sites within the UK. No credible alternative options have been identified for this waste stream at this time under the Waste Management Arrangements [Ref-36]. The strategy to dispose to permitted disposal sites is consistent with the conclusions made by LLWR strategic BAT assessment whilst there is sufficient flexibility for the future operator to select disposal options based on site specific information.

5.5. Claim 5 - Minimise the Impacts on the Environment and Members of the Public from Radioactive Waste that is Disposed of to the Environment

The design of the UK ABWR has focused on reducing the amount of radioactivity in gaseous and liquid wastes that are discharged from the facility. However, where discharges of radioactivity to air and water are unavoidable, techniques have been adopted to ensure that the subsequent impacts to the environment and members of the public are ALARA.

The Arguments presented in support of this Claim are considered to demonstrate compliance with the standard BAT conditions [Ref-2]:

- Condition 2.3.2(c) The operator shall use the best available techniques in respect of the disposal of radioactive waste pursuant to the permit to dispose of radioactive waste at times, in a form, and in a manner so as to minimise the radiological effects on the environment and members of the public.

This is also considered to fulfil the following requirement of the P&ID [Ref-1]:

- Minimising the impact of those discharges on people, and adequately protecting other species.

The UK ABWR design contains a range of features that contribute to the substantiation of this Claim including:

- Minimising the impact of discharges to the environment by means of optimising the design and operation of any discharge outlets.

In developing the Arguments presented to demonstrate the validity of Claim 5 REPs have been taken into account. The following REPs are considered to be specifically relevant to this Claim:

- **Principle RSMDP7** 'When making decisions about the management of radioactive substances, the best available techniques should be used to ensure that the resulting environmental risk and impact are minimised.
- **Principle RPDP1** 'All exposures to ionising radiation of any member of the public and of the population as a whole shall be kept ALARA, economic and social factors being taken into account.'
- **Principle ENDP2** 'Radiological impacts to people and the environment should be avoided and where that is not practicable minimised commensurate with the operations being carried out.'
- **Principle ENDP16** 'Best available techniques should be used in the design of ventilation systems.'
- **Principle DEDP3** 'Facilities should be designed, built and operated using the best available techniques to minimise the impacts on people and the environment of decommissioning operations and the management of decommissioning wastes.'

5.5.1.Argument 5a: Gaseous Discharge System - Main Stack

Significant efforts have been expended to remove radioactivity from gaseous wastes generated in the UK ABWR but some radioactivity will be discharged to the environment. The location of the discharge points will have a bearing on the impact to members of the public and the environment, however, as the gaseous discharges will be continuous, timing of discharges is not a consideration for minimising the impact.

The majority of the gaseous radioactive waste will be discharged via the main stack which is located on the roof of the R/B. The main stack receives gaseous wastes from the OG and HVAC and TGS system. The location of the main stack has been selected because of its proximity to the systems that feed into it, the height provided by the R/B of which it is part and the structural strength of the R/B (5.5.1.1 Evidence: Main Stack – Location). Modelling of the impacts associated with discharges will be undertaken to demonstrate the relationship between the height of the stack and the impact to members of the public from the radioactivity of the waste that is discharged (5.5.1.2 Evidence: Stack Height Determination – Main stack) (5.5.1.3 Evidence: Gaseous Discharge – Dilution Factor). The assessment will also consider the costs (costs are impacted by relevant factors including seismic requirements, civil engineering and hazards) of the engineering associated with increasing stack-height and explore at what point further increases in the height of the stack is grossly disproportionate to the benefits that are realised from reductions in impacts to members of the public and the environment. Undertaking a systematic determination of stack height at the GDA stage is not possible as the actual stack height will be dependent on factors that will not be available until site specific issues have been considered such as the meteorological conditions, topography, location of surrounding buildings, location of sensitive receptors and the final site layout. However, an assumed stack height of 57m has been used at GDA for dose modelling assessments.

The design of the UK ABWR main stack includes the provision of equipment that allows for sampling of gaseous radioactive waste that is discharged to the environment [Ref-20]. The selection of the techniques for sampling and analysis of waste will be made by future operators. The future operator will also be responsible for defining the environmental monitoring programme which will allow the actual impacts on members of the public and the environment from discharges to be retrospectively determined (FA9).

5.5.1.1. Evidence: Main Stack - Location

The main stack will be located on the roof of the R/B which provides the following benefits:

- Minimises the length of duct work required to transfer gaseous radioactive waste between the source and the final discharge point; and
- It benefits from the height and structural strength provided by the R/B. The additional height contributes towards effective dispersion.

5.5.1.2. Evidence: Stack Height Determination - Main Stack

The height of the UK ABWR main discharge stack will be determined by a suitably qualified and experienced person using an appropriate tool or technique in accordance with the 'Approach to Optimisation' [Ref-3] to ensure effective dilution and dispersion of gaseous radioactive waste in order to minimise the dose to members of the public and the environment (FA12). The height will be determined, taking into account local site topography and wind patterns. At this stage the precise detailed design of the discharge stack and surrounding buildings for the UK ABWR is not known. For the purposes of GDA the main stack heights of the 4 existing Japanese ABWRs have been used to inform the stack height that has been used in the dose modelling. The main stack heights of the 4 existing Japanese ABWRs are 57m, 75m, 98m and 100m. Therefore, for the purposes of the dose modelling assessments, a proposed design stack height of 57m has been used. This was set conservatively as the shortest potential stack height for a UK ABWR [Ref-102].

The height of the UK ABWR main stack will be determined at the site specific environmental permit application stage by detailed assessment that takes account of the following:

- dispersion modelling taking into account site characteristics (site topography, wind patterns etc.);
- dose to the public and the environment;
- safety aspects:
    - off-site doses due to on-site releases in normal and accidental operation;
    - on-site doses to workers; and
    - industrial safety;
- cost aspects:
    - design of HVAC systems;
    - technical limitations; and
    - construction cost;
- planning requirements, including visual impact.

The assessment will be governed by the proportionality principle to ensure optimisation is achieved when any additional increase of the stack height would incur costs that are grossly disproportionate to the benefits provided.

5.5.1.3. Evidence: Gaseous Discharge – Dilution Factor

Gaseous radioactive waste with the highest concentration of radioactivity is subject to treatment within the OG. This waste stream has a relatively small volumetric flow compared to the combined discharge from the HVAC systems which have very low concentrations of radioactivity. Following appropriate abatement the gaseous radioactive waste from the OG is therefore subject to considerable dilution prior to being discharged to the environment via the main stack. The TGS exhaust is combined with the OG and is mixed with the HVAC extract prior to being discharged via the main stack. The TGS is therefore subject to the same dilution and subsequent dispersion that is achieved for the OG.

The volumetric flow data is based on the following:

- Reactor area HVAC: 228,285 m^3/h;
- T/B HVAC: 353,143 m^3/h;
- Radioactive waste building HVAC: 146,289 m^3/h;
- Turbine gland air and steam : 3,160 m^3/h; and
- Driving air for OG ejector: 146 m^3/h.

The HVAC, turbine gland steam and OG ejector driving air volumetric flow under normal operations is 731,023 m^3/h [Ref-71]. Along with the HVAC system, the stack also serves the OG. The volumetric flow rate from the OG via the main stack during normal operations is 40 m^3/h [Ref-16]. The dilution factor presented in Table 5.5.1.3-1 has been calculated using Calculation 5.5.1.3-1.

Dilution factor = Vf / Vi

Where;

Vf: The volume of exhaust air from the HVAC systems, turbine gland steam and OG ejector driving air (m^3/h); and

Vi: The volume of exhaust air from the OG (m^3/h).

Calculation 5.5.1.3-1 Dilution Factor Calculation

Table 5.5.1.3-1: Dilution Factor for Gaseous Radioactive Waste from the OG

Note: flow rate is normalised to the condition with temperature of 0° ambient pressure of 101.325kPa.

Volumetric flow from The HVAC, turbine gland steam and OG ejector driving air **(m^3/h)**	**Volumetric flow from OG (m^3/h)**	**Dilution factor**
731,023	40	approx. 18,000

5.5.2. Argument 5b: Liquid Effluent System

Significant efforts have been expended to remove radioactivity from liquid wastes that are generated in the UK ABWR, but some radioactivity will be discharged to the environment. The location and timing of these discharges will have a direct bearing on the impact to members of the public and the environment from operations of the UK ABWR (5.5.2.1 Evidence: Cooling Water Discharge Location).

All of the radioactive liquid effluent will be discharged with the cooling water that is used to condense the steam and to provide emergency cooling. The flow rate of cooling water through the UK ABWR will depend on the operational status of the reactor unit and will be between approximately 8,000m^3/hr and approximately 200,000m^3/h (5.5.2.2 Evidence: Liquid Effluent Discharges – Dilution). For the provision of GDA it is assumed that the generic site is coastal and the cooling will be provided by a once through sea water cooling system.

The design of the UK ABWRs liquid effluent management system allows the timing of the discharges to be controlled to take account of any prevailing environmental conditions and regulatory requirements (5.5.2.3 Evidence: Control and Management of Aqueous Discharges). The cooling water will be discharged into the sea adjacent to the location of the UK ABWR. The exact position of the discharge point will be defined by the future operator of the NPP (FA13). The models used to determine the impacts to members of the public and the environment from liquid effluent discharges reflect the generic site description and are insensitive to the position of the discharge provided it is within 10km of the site and within 5km of the shore (5.5.2.4 Evidence: GDA Dose Modelling). Assessment of the discharges using these models has demonstrated that the impacts of the discharges will be very low.

The design of the UK ABWRs liquid effluent management system includes equipment that allows the effluent to be sampled prior to discharge to the environment. This will allow the operator to undertake analysis and to confirm that the characteristics of the waste conform with any specific limitations and conditions within the site's environmental permit. The design does not currently include equipment that allows the effluent to be sampled during the discharge. The Approach to Sampling and Monitoring report assesses the equipment required to take samples of the discharge and allow the operator to provide a true and accurate record of the radioactivity discharged to the environment [Ref-20]. The selection of the techniques for sampling and analysis of waste shall be made by the future operator of the NPP. The future operator shall also be responsible for defining the environmental monitoring programme which will allow the actual impacts on members of the public and the environment from discharges to be retrospectively determined (FA9).

5.5.2.1. Evidence: Cooling Water Discharge Location

The cooling water, along with any liquid waste from the HCW, LD and CAD, will be discharged into the sea adjacent to the location of the UK ABWR.

Radioactive liquid effluents, from the HCW, LD and CAD, are discharged along with the stations cooling water to achieve a high degree of dilution. This is achieved by selecting the means of final discharge into the sea at a sufficient distance from the shore to ensure that the radioactive effluents, already diluted by cooling water, are further diluted and dispersed in the marine environment.

The exact position of the discharge point will be defined by the future operator of the NPP taking into account dilution and dispersion assessments based on site specific characteristics (FA13).

5.5.2.2. Evidence: Liquid Effluent Discharges – Dilution

The discharge rate of cooling water through the UK ABWR will depend on the operational status of the reactor unit and will be between approximately 9,000 m^3/h and approximately 200,000 m^3/h. Radioactive aqueous waste from the HCW, LD system and CAD will also be discharged along with the non-radioactive cooling water. The discharge rate of radioactive aqueous waste is calculated to be approximately 4,000 m^3/year based on design basis calculations. The radioactive aqueous waste is treated by the systems described in Section 5.2.5 (Argument 2f; Configuration of Liquid Management Systems) and HCW should not be discharged except as needed to maintain the plant water balance. The radioactive aqueous waste will therefore be subject to a very large amount of dilution prior to being discharged into the environment.

5.5.2.3. Evidence: Control and Management of Aqueous Discharges

Management of discharges will take into account tidal, hydrological and geomorphological features and other factors that could affect the dilution and dispersion of radioactive liquid effluents. The arrangements will be defined by the future operator following appropriate assessment (FA13).

5.5.2.4. Evidence: GDA Dose modelling

The Environment Agency has developed a simple and cautious assessment methodology of the critical group dose for the prospective assessment of public doses [Ref-93], [Ref-94] , known as the Initial Radiological Assessment Tool (IRAT) methodology. As dictated by RGP, the IRAT methodology supports a staged approach to the dose assessment process:

- Stage 1: uses the IRAT methodology and standard generic parameters which enable a cautious assessment of the radiological impact of discharges.

  If the assessed dose is > 20μSv/y, then proceed to Stage 2.

- Stage 2: uses the IRAT methodology and refined data with more realistic parameters.

  If the assessed dose is > 20μSv/y, then proceed to Stage 3.

- Stage 3: uses more detailed site-specific data. Stage 3 does not use the IRAT methodology but more detailed codes (in the case of Hitachi-GE, PC CREAM 08®). The computer code PC CREAM 08® comprises of a number of modules that predict the transfer of radionuclides in the environment.

At Step 3 of the GDA the model input parameters have been defined based on generic site and release conditions [Ref-42]. The results of the modelling demonstrate that the discharges from the UK ABWR at the generic site would not threaten any dose limits or constraints.

6. Radionuclide Specific BAT Route Map

Table 6-1 and Table 6-2 provide a list of the main radionuclides for gaseous and liquid releases. For each radionuclide, detail is provided on its production mechanism, route to the environment, the techniques used to eliminate or reduce generation at source and the techniques used to minimise the impacts on the environment. The techniques are described in the Claims, Arguments and Evidence set out within this report. The expected Annual Discharges are also provided for information (further details on discharges are provided in 'Quantification of Discharges and Limits' [Ref-10]). Table 6-3 maps the same information for each solid waste category. It provides links to the relevant Claims, Arguments and Evidence and also provides a summary of how each waste is characterised and subsequently managed.

Table 6-1: Summary table of Main Radionuclides for Gaseous releases

Radio-nuclide	Annual Discharge (Bq/y)	Production mechanism	Route to the environment	Technique to eliminate or reduce the generation at source (Claim 1)	Technique to minimise the impacts on the environment (Claim 2-5)
Argon-41	1.8E+12	Argon-40(n,g) Argon-41 Activation of entrained atmospheric Ar in coolant.	Activation of coolant → Migration into steam → Separation at condenser → Discharge via stack.	Minimisation of leaks (Argument 1j) and the air leakage into the main condenser.	Off-gas treatment system charcoal delay beds (Argument 2a and 2b). Discharge at height via main stack (Argument 5a).
Krypton-85	1.0E+08	FPs from fuel and structural uranium. Radioactive noble gases are formed by fission. They are usually confined within the fuel but in the event of fuel leaks, they can pass into the coolant via defects in the fuel cladding. Their presence in the coolant is also due to the occurrence of traces of uranium ("tramp" uranium) on the surface of fuel assemblies following the manufacturing process.	Migration into reactor water (direct or through pin failure) → 100 percent migration into steam → Separation at condenser → Discharge from stack via OG.	Minimise fuel cladding failures (grid-to-rod fretting, corrosion and crud, debris, PCI and manufacturing QA) (Argument 1a). High standards of fuel design and fabrication (Argument 1a). Minimise "tramp uranium" (Argument 1a). Minimisation of crud formation and optimal water chemistry (Argument 1f). An efficient anti debris device is provided for fuel assemblies (Argument 1a). The fuel performance - minimising the number of fuel assemblies used minimises the probability for cladding leakage of FPs into the coolant (Argument 1c). Identifying and isolating fuel leaks (Argument 1d). Minimise leaks (Argument 1j). Design change to prevent discharge of reactor steam (Argument 2e).	Off-gas treatment system and charcoal delay beds (Argument 2a and 2b). Discharge at height via main stack (Argument 5a).
Krypton-85m	2.3E+09				
Krypton-87	2.3E+03				
Krypton-88	1.8E+08				
Krypton-89	0.0E+00				
Xenon-131m	1.4E+08				
Xenon-133	1.0E+10				
Xenon-133m	1.7E+06				
Xenon-135	1.7E-11				
Xenon-135m	0.0E+00				
Xenon-137	0.0E+00				
Xenon-138	0.0E+00				
Iodine-131	1.9E+08	FPs from fuel, structural uranium. Iodine isotopes are formed in the fuel by fission and can escape into the reactor coolant water via fuel defects. Also, like other FPs, small quantities are produced from uranium contamination on fuel surface ("tramp" uranium) within the reactor which can also be found in the coolant.	Migration into reactor water (direct or through pin fracture) → Partial migration into steam → Separation at condenser → Discharge via stack via OG(negligible). Discharge of volatile iodine in aqueous stream via HVAC system. Discharge of iodine in gaseous radioactive waste discharged to main stack from the gland steam exhauster.	Minimise fuel cladding failures (grid-to-rod fretting, corrosion and crud, debris, PCI, and manufacturing upsets) (Argument 1a). High standards of fuel design and fabrication (Argument 1a). Minimise "tramp uranium" (Argument 1a). Minimisation of crud formation and optimal water chemistry (Argument 1f). An efficient anti debris device is implemented for fuel assemblies (Argument 1a). The fuel performance - minimising the number of fuel assemblies used minimises the probability for cladding leakage of FPs into the coolant (Argument 1c),	Off-gas treatment system and charcoal delay beds (Argument 2a and 2b). Discharge at height via main stack (Argument 5a).

Radio-nuclide	Annual Discharge (Bq/y)	Production mechanism	Route to the environment	Technique to eliminate or reduce the generation at source (Claim 1)	Technique to minimise the impacts on the environment (Claim 2-5)
				Identifying and isolating fuel leaks (Argument 1d). Minimise leaks (Argument 1j).	
Strontium-89	9.9E+03	FPs from fuel, structural uranium. Isotopes of strontium are formed as a result of fission. They are usually confined in the fuel but, in the event of fuel leaks, they can pass into the coolant via defects in the fuel cladding. Their presence in the coolant is also due to the occurrence of traces of uranium ("tramp" uranium) that can never be completely removed on new fuel assemblies following the manufacturing process.	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.	Minimise fuel cladding failures (grid-to-rod fretting, corrosion and crud, debris, PCI, and manufacturing upsets) (Argument 1a). High standards of fuel design and fabrication (Argument 1a). Minimise "tramp uranium" (Argument 1a). Minimisation of crud formation and optimal water chemistry (Argument 1f). An efficient anti debris device is implemented for fuel assemblies (Argument 1a). The fuel performance - minimising the number of fuel assemblies used minimises the probability for cladding leakage of FPs into the coolant (Argument 1c). Identifying and isolating fuel leaks (Argument 1d). Minimise leaks (Argument 1j).	Filters to remove particulate material (Argument 2d). Discharge at height via main stack (Argument 5a).
Strontion-90	6.3E+02				
Cesium-137	1.4E+03	FPs from fuel, structural uranium.			
Cobalt-60	3.7E+04	Cobalt-59 (n,g) cobalt-60. 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) Aerosol generation in tanks, pools and release via HVAC system. Discharge of evaporated particulate in aqueous stream via TGS system.	Minimisation of crud formation and optimal water chemistry (Argument 1f). Specification of low cobalt content materials (Argument 1g). Minimise leaks (Argument 1j).	Filters to remove particulate material (including filters on the HVAC) (Argument 2d). Discharge at height via main stack (Argument 5a).
Tritium	2.7E+12	Ternary fission in fuel Boron-10 (n,2a).Tritium from boron in control rods. Hydrogen-2 (n,g) H-3 from Hydrogen-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. Adventitious discharges from steam leaks → Discharge via HVAC of aqueous vapour. Evaporative losses from sources of tritiated water→ Discharge of aqueous vapour via HVAC system. Discharge of tritium in gaseous radioactive waste discharged to main stack from the gland steam exhauster.	No boron usage in the water chemistry (Argument 1b). Use of hafnium control rods (Argument 1b). Use of gadolinium as a burnable poison rather than boron (Argument 1b). Minimise fuel cladding failures (grid-to-rod fretting, corrosion and crud, debris, PCI, and manufacturing upsets) (Argument 1a). High standards of fuel design and fabrication (Argument 1a). Minimisation of crud formation and optimal water chemistry (Argument 1f). An efficient anti debris device is implemented for fuel assemblies (Argument 1a). The fuel performance - minimising the number of fuel assemblies used minimises the probability for cladding leakage of FPs into the coolant (Argument 1c).	Gaseous tritium present within the off-gas is removed by the off-gas recombiner and off-gas condenser. The off-gas Recombiner recombines hydrogen and oxygen and the off-gas condenser cools and condenses the hydrogen depleted off-gas to separate any moisture and return it to the main condenser. Following treatment by these two components of the OG the hydrogen concentration is minimised in the off-gas. As tritium is a hydrogen compound; the performance of the off-gas recombiner and off-gas condenser therefore also removes tritium from the off-gas. The hydrogen and therefore any tritium is converted to water and is returned to the CST where it is reused within the plant. (Argument 2a).

Radio-nuclide	Annual Discharge (Bq/y)	Production mechanism	Route to the environment	Technique to eliminate or reduce the generation at source (Claim 1)	Technique to minimise the impacts on the environment (Claim 2-5)
				Identifying and isolating fuel leaks (Argument 1d) Minimise leaks (Argument 1j). Condense 98 percent of TGS steam minimising residual steam discharge and tritium (Argument 2e).	Discharge at height via main stack (Argument 5a).
Carbon-14	9.1E+11	Neutron activation of Nitrogen-14 and Oxygen-17results in Carbon-14 both from fuel and reactor water. Another minor mechanism contributing to Carbon-14 is the reaction Carbon-13 (n, γ) -> Carbon-14, which occurs due to the presence of dissolved carbon in the coolant.	Carbon-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.	None The principle source of Carbon-14 is the thermal neutron reaction with Oxygen-17 in the reactor coolant water (H_2O). Therefore, there are no measures for reducing the generation of Carbon-14.	None. (Argument 2a). Discharge at height via main stack (Argument 5a).

Table 6-2: Summary table for Main Radionuclides for liquid releases

Radio-nuclide	Annual Discharge (Bq/y)	Production mechanism	Route to the environment	Technique to eliminate or reduce the generation at source (Claim 1)	Technique to minimise the impacts on the environment (Claim 2-5)
Strontium-89	2.2E+03	FPs 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	Minimise fuel cladding failures (grid-to-rod fretting, corrosion and crud, debris, PCI, and manufacturing upsets) (Argument 1a). High standards of fuel design and fabrication (Argument 1a). Minimise "tramp uranium" (Argument 1a). Minimisation of crud formation and optimal water chemistry (Argument 1f). An efficient anti debris device is implemented for fuel assemblies (Argument 1a). The fuel performance - minimising the number of fuel assemblies used minimises the probability for cladding leakage of FPs into the coolant (Argument 1c). Identifying and isolating fuel leaks (Argument 1d). Minimise leaks (Argument 1j).	CUW system (Argument 1h) LD pre-filter LD activated carbon adsorption tower LD filter HCW evaporator HCW demineraliser (Argument 2f, Argument 2h and 2i)
Strontium-90	1.1E+03				
Iodine-131	3.5E+04	FPs 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		
Cesium-137	1.6E+03	FPs 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	Minimise fuel cladding failures (grid-to-rod fretting, corrosion and crud, debris, PCI, and manufacturing upsets) (Argument 1a). High standards of fuel design and fabrication (Argument 1a). Minimise "tramp uranium" (Argument 1a). Minimisation of crud formation and optimal water chemistry (Argument 1f). An efficient anti debris device is implemented for fuel assemblies	CUW system (Argument 1h) LD pre-filter LD AC filter LD activated carbon adsorption tower HCW evaporator

Radio-nuclide	Annual Discharge (Bq/y)	Production mechanism	Route to the environment	Technique to eliminate or reduce the generation at source (Claim 1)	Technique to minimise the impacts on the environment (Claim 2-5)
			Discharge via HCW sample tank- occasional	(Argument 1a). The fuel performance - minimising the number of fuel assemblies used minimises the probability for cladding leakage of FPs into the coolant (Argument 1c). Identifying and isolating fuel leaks (Argument 1d). Minimise leaks (Argument 1j).	HCW demineraliser. (Argument 2f, Argument 2h and 2i).
Cobalt-60	2.0E+05	Cobalt-59 (n,g) Cobalt-60 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	Minimisation of crud formation and optimal water chemistry (Argument 1f). Specification of low cobalt content materials (Argument 1g). Minimise leaks (Argument 1j).	CUW system (Argument 1h) LD pre-filter LD filter LD activated carbon adsorption tower HCW evaporator HCW demineraliser (Argument 2f, Argument 2h and 2i)
Tritium	2.0E+11	Ternary fission in fuel. Boron-10 (n,2a) Tritium (from boron in control rods). Hyrogen-2 (n,g) tritium (from Hydrogen-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	No boron usage in the water chemistry (Argument 1b). Use of hafnium control rods (Argument 1b). Use of gadolinium as a burnable poison rather than boron (Argument 1b). Minimise fuel cladding failures (grid-to-rod fretting, corrosion and crud, debris, PCI and manufacturing upsets) (Argument 1a). High standards of fuel design and fabrication (Argument 1a). Minimisation of crud formation and optimal water chemistry (Argument 1f). An efficient anti debris device is implemented for fuel assemblies (Argument 1a). The fuel performance - minimising the number of fuel elements used minimises the probability for cladding leakage of FPs into the coolant (Argument 1c). Identifying and isolating fuel leaks (Argument 1d). Minimise leaks (Argument 1j).	No abatement (Argument 2f)

Table 6-3: Summary table of solid waste arisings

Disposal volumes for each waste stream are given in Appendix A of E4 Radioactive Waste Management Arrangements [Ref-36]

Waste category	Key radionuclides and specific activity [Ref-36]	Description	Source		Technique to eliminate or reduce the generation at source	Technique to minimise the impacts of disposal on the environment (Claims 2-5)	Characterisation	Waste management route
VLLW	Dependent upon operational source; mainly steel activation products.	Miscellaneous combustible: paper, polythene, cloth etc.	Maintenance operations		Prevent or reduce the generation of FPs, activation products and CPs which subsequently lead to the generation of combustible and non combustible VLLW during maintenance activities (5.1 Claim 1: Eliminate or Reduce the Generation of Radioactive Waste).	Segregation of waste to ensure the optimised treatment, storage and disposal option is selected (5.3.2.1 Evidence: Segregation of Waste). Minimising the number of operator visits into RCAs and reducing the volume of consumables that are taken into RCAs reduces the potential to generate maintenance wastes (5.3.2.2. Evidence: Locate Offices Outside of Controlled Areas). Minimising the amount of maintenance equipment and tools that are taken into RCA (5.3.2.3. Evidence: Storage Facilities for Tools and Other Maintenance Equipment). Provision of facilities to undertake the characterisation, sorting, treatment and storage of waste prior to consignment to an appropriately permitted waste management service supplier (5.4.1. Argument 4a: Provision of Waste Management Facilities). Effective preventative maintenance schedules; predict, prepare and avoid (where practicable) leaks and spillages and associated clean-up activities (5.3.2.4. Evidence: Maintenance Philosophy and [Ref-36]). Volume reduction treatment processes (5.3.3.2. Evidence: Incineration and 5.3.3.3. Evidence: Solid Waste Compaction). Decontamination where practicable to reduce waste classification and / or aid onward treatment and disposal [Ref-36].	Anticipated waste characteristics determined at work planning phase (data collection to support history and provenance arguments). Measurement at source. Potential for sampling and off-site detailed characterisation. Gamma assay systems installed in solid LLW facility. Application of source area / plant fingerprint [Ref-36].	Incineration (5.3.3.2. Evidence: Incineration and 5.4.2.1. Evidence: Waste Treatment Techniques and Disposal Routes).

Waste category	Key radionuclides and specific activity [Ref-36]	Description	Source		Technique to eliminate or reduce the generation at source	Technique to minimise the impacts of disposal on the environment (Claims 2-5)	Characterisation	Waste management route
Dry-Solid LLW	iron-55, cobalt-60, zinc-65, manganese-54, cesium-137, strontium-90, antimony-125	HVAC filters.	HVAC system		Prevent or reduce the generation of FPs, activation products and CPs which could then be filtered by the HVAC system leading to the contamination of HVAC filters (5.1 Claim 1: Eliminate or Reduce the Generation of Radioactive Waste).	HEPA filters will be changed, where practicable, based on performance determined using continuous measurement of differential pressures or as a result of manufacturer's guidance (5.2.4. Argument 2d: Filtration of Airborne Particulate Matter and 5.3.2.4. Evidence: Maintenance Philosophy). HVAC filters will be segregated from other waste streams to ensure appropriate maintenance (5.3.2.1. Evidence: Segregation of Waste). Volume reduction treatment processes (5.3.3.2. Evidence: Incineration and 5.3.3.3. Evidence: Solid Waste Compaction).	Same as for VLLW.	Compaction as filters are not incinerable (5.4.2.1. Evidence: Waste Treatment Techniques and Disposal Routes and BAT options assessment report [Ref-43]).
	iron-55, cobalt-60, manganese-54, nickel-63	Miscellaneous combustible: paper, polythene, cloth, LCW filter membrane, spent activated carbon.	As for VLLW combustible plus LD and LCW.		Prevent or reduce the generation of FPs, activation products and CPs which subsequently lead to the generation of combustible LLW during maintenance activities (5.1 Claim 1: Eliminate or Reduce the Generation of Radioactive Waste).	See VLLW 2 - 3 years decay storage of bead activated carbon from LD system (5.2.9. Argument 2j: Radioactive Decay of Solid and Liquid Wastes).	Same as for VLLW.	Incineration (5.4.2.1. Evidence: Waste Treatment Techniques and Disposal Routes and BAT options assessment report [Ref-43]).
	iron-55, cobalt-60, magnesium-54	Recyclable metals.			Prevent or reduce the generation of FPs, activation products and CPs which subsequently lead to the generation of metal LLW during maintenance activities (5.1 Claim 1: Eliminate or Reduce the Generation of Radioactive Waste)	See VLLW.	Same as for VLLW.	Off-site recycling (5.4.2.1. Evidence: Waste Treatment Techniques and Disposal Routes and BAT options assessment report [Ref-43])
	iron-55, cobalt-60, magnesium-54	Non-combustible and non-compactable waste (including metals unsuitable for recycling and CF filters).	As for VLLW non-combustible plus CF system.		Prevent or reduce the generation of FPs, activation products and CPs which subsequently lead to the generation of non-combustible LLW during maintenance activities (5.1 Claim 1: Eliminate or Reduce the Generation of Radioactive Waste).	See VLLW.	Same as for VLLW.	Disposal at an appropriately permitted site (5.4.2.1. Evidence: Waste Treatment Techniques and Disposal Routes) E.g. LLWR (BAT options assessment report [Ref-43]).

Waste category	Key radionuclides and specific activity [Ref-36]	Description	Source		Technique to eliminate or reduce the generation at source	Technique to minimise the impacts of disposal on the environment (Claims 2-5)	Characterisation	Waste management route
Wet -solid LLW	iron-55, cobalt-60, magnesium-54, nickel-63	Organic bead demineraliser resin (condensate, LCW, HCW), HCW evaporator sludge, granular activated carbon from LD system.	Condensate, LCW, HCW, LD.		Prevent or reduce the generation of FPs, activation products and CPs which subsequently enter liquid systems and require treatment leading to the generation of resin, sludge and granular activated carbon waste (5.1 Claim 1: Eliminate or Reduce the Generation of Radioactive Waste). Before start-up, removing crud prior to activation reduces the radioactivity deposited on the demineraliser resins (5.1.5.3. Evidence: Water Conditioning).	Segregated between source systems (5.2.5. Argument 2f: Configuration of Liquid Management Systems). Segregation of waste to ensure the optimised treatment, storage and disposal option is selected (5.3.2.1 Evidence: Segregation of Waste). Cementation, in batch campaigns, prior to disposal (5.4.1.3. Evidence: Waste Treatment Facilities). Replacement of pre-coated filters to HFF or pleated filters (5.3.1.5 Evidence: Replacement of Pre-coated Filters). During an outage the CD is isolated and is stored in demineralised water to prevent degradation of the resin (5.1.5.3. Evidence: Water Conditioning). Selection of resin media that can be suitably disposed (5.1.8.6. Evidence: Demineraliser Media). Allowing solid and liquid radioactive waste to undergo radioactive decay before disposing of it to the environment or another premises will reduce the amount of radioactivity that is disposed of in the waste (5.2.9. Argument 2j: Radioactive Decay of Solid and Liquid Wastes).	Anticipated waste characteristics determined at work planning phase (data collection to support history and provenance arguments). Sampling and off-site detailed characterisation of raw waste (including cement formulation trials). In-line measurements in the processing plant. Monitoring of final waste package and application of radioactivity fingerprint [Ref-36].	Disposal at an appropriately permitted site (5.4.2.1. Evidence: Waste Treatment Techniques and Disposal Routes) E.g. LLWR (BAT options assessment report [Ref-75]).

Waste category	Key radionuclides and specific activity [Ref-36]	Description	Source		Technique to eliminate or reduce the generation at source	Technique to minimise the impacts of disposal on the environment (Claims 2-5)	Characterisation	Waste management route
Dry-Solid ILW	cobalt-60, nickel-63 (californium-252 in neutron source)	Activated Metals: control rods, reactor components (e.g. neutron source unit).			The design of the UK ABWR has evolved to reduce the quantities of solid radioactive waste that will be generated during its life-cycle (5.3.1 Argument 3a: Design to Minimise the Volumes of Operational and Decommissioning Waste Arisings). Use of hafnium control rods which have a longer operational life and therefore require less frequent disposal (5.1.2. Argument 1b: Reactivity Control). Implementation of commissioning, start-up, shutdown and outage processes to prevent the deposition of radioactivity on reactor components which will become waste during maintenance and decommissioning (5.1.5 Argument 1e: Commissioning, Start-Up, Shutdown and Outage Procedures). Selection of materials and water chemistry to reduce the activation of metals (5.1.6. Argument 1f: Water Chemistry and 5.1.7. Argument 1g: Specification of Materials).	Segregation of waste at source and separation (LLW: ILW) after decay storage (5.3.2.1 .Evidence: Segregation of Waste and 5.4.1.2. Evidence: Segregation and Sorting Facilities). Provision of a dedicated facility to process and treat Dry-Solid ILW (5.4.1 Argument 4a: Provision of Waste Management Facilities). Decay storage in dry casks to reduce activity levels (5.2.10. Argument 2j: Radioactive Decay of Solid and Liquid Wastes). Size reduction in order to aid optimal disposal (5.3.3.1. Evidence: Size Reduction of Control Rods). Optimised disposal (5.4.2.Argument 4b: Optimal Disposal Route Selection, 5.4.4.Argument 4d: Disposability Assessments for Higher Activity Wastes and 5.4.5 Argument 4e: Compatibility of Existing UK Waste BAT Studies).	Anticipated waste characteristics determined at work planning phase (data collection to support history and provenance arguments). Measurement and assay following decay storage [Ref-36].	Disposal at GDF (5.4.4.2. Evidence: Disposability Assessment – Intermediate Level Waste and Options for the management of Dry Solid ILW [Ref-75]).

Waste category	Key radionuclides and specific activity [Ref-36]	Description	Source		Technique to eliminate or reduce the generation at source	Technique to minimise the impacts of disposal on the environment (Claims 2-5)	Characterisation	Waste management route
Wet-Solid ILW	iron-55, cobalt-60, magnesium-54, zinc-65	Organic powder demineraliser resin (CUW and FPC systems), sludge (crud) from CF and LCW filters.	CUW, FPC, CF & LCW.		Prevent or reduce the generation of FPs, activation products and CPs which subsequently enter liquid systems and require treatment leading to the generation of resin, sludge and granular activated carbon waste (5.1 Claim 1: Eliminate or Reduce the Generation of Radioactive Waste). Before start-up, removing crud prior to activation reduces the radioactivity deposited on the demineraliser resins (5.1.5.3. Evidence: Water Conditioning).	See wet solid LLW.	Anticipated waste characteristics determined at work planning phase (data collection to support history and provenance arguments). Sampling and off-site detailed characterisation of raw waste (including cement formulation trials). In-line measurements in the processing plant. Monitoring of final waste package and application of radioactivity fingerprint [Ref-36].	Disposal at GDF (5.4.4.2. Evidence: Disposability Assessment – Intermediate Level Waste).
Fuel	FPs, activation products and actinides	Fuel assemblies of GE14 design: uranium dioxide pellets within Zircaloy cladding; fuel rods held in bundles.	Fuel		The efficiency with which the nuclear fuel is used in the UK ABWR and the frequency with which it is changed will influence the amount of SF and HAW that is generated during operations (5.1.3. Argument 1c: Efficiency of Fuel Use). The generation of fuel waste is inevitable however there are a number of practices which ensure subsequent optimal handling, treatment and disposal (5.1.1. Argument 1a: Design, Manufacture and Management of Fuel).	Segregation of fuel from other waste streams (5.3.2.1. Evidence: Segregation of Waste). Decay storage in SFP followed by dry cask storage in SFIS [Ref-36].	Calculated waste characteristics based on materials and operations.	Disposal at GDF (5.4.4.1 Evidence: Disposability Assessment – Spent Nuclear Fuel).

7. Forward Action Plan

Table 7-1: Forward Action Plan

No.	Section Reference	Action	Delivery phase
1.	5.1.8	To support the demonstration that (from 5.1.8 Argument 1h: Recycling of Water within Steam Circuit) performance of systems deemed to be BAT perform as expected and have therefore been optimised, a future operator shall undertake performance monitoring of the following systems during commissioning: • Condensate Water Clean-Up System; • CUW; • FPC and SPCU; and • LCW.	Commissioning
2.	5.1.8	An assessment shall be undertaken to determine BAT for the selection of demineraliser resins to support the full substantiation of argument 5.1.8 Argument 1h: Recycling of Water within Steam Circuit.	Future Operator
3.	5.2.1.3	Undertake a BAT assessment of carbon-14 abatement techniques including Alkaline Scrubbing (e.g. determine if available evidence challenges the argument that the development and implementation of an abatement technique is grossly disproportionate).	Future operator
4.	5.2.2.1	Commissioning data shall be provided to support the design basis calculations currently being used to substantiate the argument that the delay period provided by the off-gas delay beds is BAT (5.2.2 Argument 2b: Delay Beds for Noble Gases and iodine).	Commissioning
5.	5.3.4	Undertake a BAT assessment of waste management techniques post-GDA taking into account site specific factors including the proximity principle.	Future operator
6.	5.4.1 5.4.2 5.4.2.1	Undertake BAT assessments to support the specification and selection of equipment to be used in the radioactive waste management building (5.4.1.3 Evidence: Waste Treatment Facilities).	Future operator
7.	5.4.2 5.4.3.1 5.4.5.1	Undertake a BAT assessment of waste management routes taking into account site specific factors including the proximity principle and other relevant factors to fully substantiate 5.4.2 Argument 4b: Optimal Disposal Route Selection.	Future operator
8.	5.3.4.1 5.4.1.1 5.4.1.3	The management of waste, the final waste route and the quantity of waste to be consigned will be determined through the application of BAT by the	Future operator

No.	Section Reference	Action	Delivery phase
	5.4.3 5.4.4	future operator.	
9.	5.5.1 5.5.2	A future operator shall select the techniques for environmental sampling and determine the environmental monitoring programme.	Future operator
10.	5.1.7.1	A future operator shall assess the cobalt content of steels based on availability and cost from available suppliers.	Future operator
11.	5.4.1.4 5.4.4.1	A future operator shall demonstrate BAT when selecting their plans for packaging, storage and disposal of SF.	Future operator
12.	5.5.1.2	A future operator shall determine the optimal stack height.	Future operator
13.	5.5.2 5.5.2.1 5.5.2.3	A future operator shall determine the management and arrangements for aqueous discharges.	Future operator
14.	5.2.9.2	A future operator shall assess and define the decay storage timescales.	Future operator
15.	5.1- 6	Management arrangements will be developed to ensure that BAT is considered through the lifecycle of the project; from design to decommissioning.	Future operator

8. Conclusion

The Claims and Arguments for the UK ABWR Demonstration of BAT have been developed using information held by Hitachi-GE. The evidence provided within this Step 3 submission reflects the information used to develop the Claims and Arguments. Gaps and uncertainties identified during the development of the arguments have been subject to additional assessment or recorded as FA's to ensure that they are closed out at the most appropriate time in the project lifecycle.

Collectively the Claims, Arguments, Evidence model will support the demonstration that BAT has been applied to the UK ABWR design, allowing examination and challenge and where applicable identifying key gaps or uncertainties. During evidence gathering Hitachi-GE have consistently asked the questions:

- Can anything else be done to reduce activity of discharges, minimise volumes of solid waste or reduce impacts from discharges?
- Is the time, trouble and money associated with implementing changes grossly disproportionate to the potential benefits gained?

The demonstration of BAT is an iterative process that feeds back to the design; if during the process any areas of insufficient Evidence remain, design changes may be made to support the application of BAT. Hitachi-GE believe that the arguments set out in this Demonstration of BAT report and the substantiation provided by the existing evidence base combined with the resolution of outstanding FA's shall demonstrate that the UK ABWR has been optimised in accordance with those requirements of the P&ID that require the application of BAT.