Office for Nuclear Regulation

New Reactors Division

Step 4 Assessment of Radiological Protection for the UK Advanced Boiling Water Reactor

Assessment Report: ONR-NR-AR-17-021
Revision 0
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EXECUTIVE SUMMARY

Hitachi-GE Nuclear Energy Ltd is the designer and GDA Requesting Party for the United Kingdom Advanced Boiling Water Reactor (UK ABWR). Hitachi-GE commenced Generic Design Assessment (GDA) in 2013 and completed Step 4 in 2017.

This assessment report is my Step 4 assessment of the Hitachi-GE UK ABWR reactor design in the area of Radiological Protection.

The scope of the Step 4 assessment is to review the safety, security and environmental aspects of the UK ABWR in greater detail, by examining the evidence, supporting the claims and arguments made in the safety documentation, building on the assessments already carried out for Step 3. In addition I have provided a judgement on the adequacy of the radiological protection information contained within the Pre-Construction Safety Report (PCSR) and supporting documentation.

My assessment conclusion is:

- Hitachi-GE has provided sufficient evidence to demonstrate radiation exposures from routine and non-routine operations are below the Basic Safety Level and generally are less than the Basic Safety Objective for workers, others on the generic site and the general public. Evidence suggests that exposures have therefore been reduced so far as is reasonably practicable by design. Where this is not the case, Hitachi-GE has identified that further work is required.
- All Regulatory Observations related to Radiological Protection have been successfully closed.
- Hitachi-GE has provided sufficient evidence to demonstrate that radiological and contamination zoning is adequate to meet the requirements of GDA.
- Hitachi-GE has provided sufficient evidence to demonstrate shielding design is adequate to meet the requirements of GDA and as stated above, ensure exposures meet the required standards.
- Hitachi-GE has provided sufficient evidence to demonstrate the design for contamination control and management of exposure to high dose and dose rate areas is sufficiently mature to meet the required standards.
- I am content the radiological protection assessment has been completed and I have identified a number of assessment findings which will need to be completed by the site licensee as part of the licensing and safety case development for any site specific phase.
- The UK ABWR has successfully completed Step 4 GDA assessment.

My judgement is based upon the following factors:

- Provision of adequate claims, arguments and evidence published in the PCSR and supporting reports and references.
- Assessment of these claims, arguments and evidence against the relevant standards, published by ONR, International Atomic Energy Agency, Western European Nuclear Regulators' Association and Health and Safety Executive, and review against relevant good practice.
- Interactions with Hitachi-GE over the period of engagement and assessment.

The following matters remain, which are for a future licensee to consider and take forward in its site-specific safety submissions. These matters do not undermine the generic safety submission but require licensee input/decision at a specific site.

Through my assessment I have identified eight assessment findings focused around a number of areas including:

- Optimisation of design for decommissioning to ensure risk to workers is controlled so far as is reasonably practicable and demonstrate doses to workers will be as low as is reasonably practicable.
- Optimisation of the design of fuel route activities relating to Reactor Pressure Vessel opening and closing sequences to ensure risks are reduced so far as is reasonably practicable and doses are as low as is reasonably practicable including:
  - Reactor Pressure Vessel nut and stud release, removal, tensioning and de-tensioning methodology including equipment design with consideration of multi-stud tooling
  - Design of features for water level control
  - Assessment of activities to remove where reasonably practicable duplicate or repeated tasks

- Development of the design to:
  - Minimise the potential for loss of containment in the form of chronic leakage
  - Identify leakages through provision of suitable monitoring
  - Provide suitable recovery options for leakage

- Consideration of wider Heating, Ventilation and Air Conditioning (HVAC) design to ensure exposures to the maintenance team are controlled so far as is reasonably practicable.

Overall, based on the samples undertaken, I am broadly satisfied that the claims, arguments and evidence laid down within the PCSR and supporting documentation submitted as part of the GDA process present an adequate safety case for the generic UK ABWR design in the area of radiological protection. For this reason I support the issue of a Design Acceptance Confirmation (DAC).

LIST OF ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
AC	Atmosphere Control System
ACOP	Approved Code Of Practice
AF	Assessment Finding
ALARA	As Low Aa Reasonably Achievable
ALARP	As Low As Reasonably Practicable
ALI	Annual Limit of Intake
AP	Activation Products
ARM	Area Radiation Monitoring
BAT	Best Available Technology
B/B	Back-up Building
BDL	Bottom Drain Line
BE	Best Estimate
BEIS	Business, Energy & Industrial Strategy
BSSD	Basic Safety Standards Directive
BSL	Basic Safety Level
BSO	Basic Safety Objective
BWR	Boiling Water Reactor
CA	Cycle Average
CLAW	Control of Lead at Work Regulations 2002
CP	Corrosion Products
CPS	Condensate Purification System
CRD	Control Rod Drive
CRO	Control Room Operator
CRUD	Chalk River Unidentified Deposits
CST	Condensate Storage Tank
CUW	Clean-Up Water System
CV	Curriculum Vitae
C&I	Control and Instrumentation
DAC	Design Acceptance Confirmation
DB	Design Basis
DBA	Design Base Analysis
DST	Deposit Source Term
DZO	Depleted Zinc Oxide
EA	Environment Agency
ECC	Emergency Control Centre
EDG/B	Emergency Diesel Generator Building

EMIT	Examination Maintenance Inspection and Testing
EOP	Emergency Operation Procedure
EPD	Electronic Personal Dosemeter
EPR16	Environmental Permitting (England and Wales) Regulations 2016
ERIC-PD	Eliminate, Reduce, Isolate, Control, Personal Protective Equipment, Discipline
EUST	End User Source Term
FP/ActP	Fission Products / Actinides Products
FLSR	Flooder System of Reactor Building
FMCRD	Fine Motion Control Rod Drive
GDA	Generic Design Assessment
HEPA	High Efficiency Particulate Air
Hitachi-GE	Hitachi-GE Nuclear Energy, Ltd
HSE	Health and Safety Executive
HSWA74	Health and Safety at Work etc. Act 1974
HVAC	Heating Ventilation and Air Conditioning
HWC	Hydrogen Water Chemistry
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
ID	Identification
IEC	International Electrochemical Commission
ILW	Intermediate Level Waste
IRR85	Ionising Radiations Regulations 1985
IRR99	Ionising Radiations Regulations 1999
ISI	In-Service Inspection
ISOE	Information System on Occupational Exposure
KK-6/7	Kashiwazaki Kariwa Unit 6 / 7
L/C	Lower Component
LEV	Local Exhaust Ventilation
LLW	Low Level Waste
LMR	Liabilities Management Regulation
LOCA	Loss of Coolant Accident
LOT	Light Oil Tank
MCNP	Monte Carlo N-Particle
MCR	Main Control Room
MDEP	Multinational Design Evaluation Programme
MHSWR99	Management of Health and Safety at Work Regulations 1999
MO	Mitigating Options

MS	Minor Shortfall
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NLR	Nuclear Liabilities Regulation
NMC	Noble Metal Chemistry
NPP	Nuclear Power Plant
NRW	Natural Resources Wales
NSEDPs	Nuclear Safety and Environmental Design Principles
OG	Off-Gas
OECD-NEA	Organisation for Economic Co-operation and Development - Nuclear Energy Agency
OLNC	On-Line NobleChem™
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
OT	Outage
PCV	Primary Containment Vessel
PCSR	Pre-Construction Safety Report
PHE-CRCE	Public Health England’s Centre for Radiation, Chemical and Environmental Hazards
PO	Power Operation
POCO	Post-Operative Clean Out
PPE	Personal Protective Equipment
PSA	Probabilistic Safety Assessment
PST	Primary Source Term
PRM	Process Radiation Monitoring
PrST	Process Source Term
R/A	Reactor Area
RCCV	Reinforced Concrete Containment Vessel
RCM	Reliability Centred Maintenance
RCV	Reactor Containment Vessel
RHR	Residual Heat Removal System
REPPIR	Radiation (Emergency Preparedness and Public Information) Regulations 2001
RGP	Relevant Good Practice
RI	Regulatory Issue
RIP	Reactor Internal Pump
RO	Regulatory Observation
RP	Requesting Party
RPA	Radiological Protection Adviser

RPV	Reactor Pressure Vessel
RQ	Regulatory Query
Rw/B	Radioactive Waste Building
SA	Severe Accident
SAMGs	Severe Accident Management Guidelines
SAPs	Safety Assessment Principles
S/B	Service Building
SD	Shut Down
SFAIRP	So Far As Is Reasonably Practicable
SFIS	Spent Fuel Interim Storage
SFP	Spent Fuel Pool
SGTS	Stand-by Gas Treatment System
SJAE	Steam Jet Air Ejector
SoDA	Statement of Design Acceptability
SPT	Suppression Pool Water Surge Tank
SQEP	Suitably Qualified and Experienced Person
SS	Spent Sludge
SSC	System, Structure (and) Component
SSM	Swedish Nuclear Authority (SSM)
SSER	Safety, Security and Environmental Report
SU	Start Up
TAG	Technical Assessment Guide
T/B	Turbine Building
TGSCC	Trans-Granular Stress Corrosion Cracking
TSC	Technical Support Contractor
TGS	Turbine Gland Steam System
TÜV SÜD	Technischer Überwachungsverein Süddeutschland
U/C	Upper Component
UK	United Kingdom
UK ABWR	United Kingdom Advanced Boiling Water Reactor
UPS	Uninterruptible Power Supply
WENRA	Western European Nuclear Regulators' Association
Zn	Zinc

TABLE OF CONTENTS

Figures

Tables

Annexes

1 INTRODUCTION

1.1 Background

1. Information on the Generic Design Assessment (GDA) process is provided in a series of documents published on our website (Refs.1, 2). The expected outcome is a Design Acceptance Confirmation (DAC) for ONR and a Statement of Design Acceptability (SoDA) for the Environment Agency (EA) and Natural Resources Wales (NRW).

2. The GDA Step 3 summary report is published on our website (**http://www.onr.org.uk/new-reactors/uk-abwr/reports/step3/uk-abwr-step-3-summary-report.pdf**). Further information on the GDA process in general is also available on our website (**http://www.onr.org.uk/new-reactors/index.htm**).

3. Hitachi-GE commenced GDA in 2013 and completed Step 4 in 2017. The Step 4 assessment is an in-depth assessment of the safety, security and environmental evidence. Through the review of information provided to ONR, the Step 4 process should confirm that Hitachi-GE:

- Has properly justified the higher-level claims and arguments.
- Has progressed the resolution of issues identified during Step 3.
- Has provided sufficient detailed assessment to allow ONR to come to a judgment of whether a DAC can be issued.

4. During the step 4 assessment I have undertaken a detailed assessment, on a sampling basis of the safety case evidence. I have not reviewed security evidence in any significant depth. The full range of items that might form part of the assessment is provided in ONR's GDA Guidance to Requesting Parties (http://www.onr.org.uk/new-reactors/ngn03.pdf) (Ref. 1). These include:

- Consideration of issues identified in Step 3.
- Judging the design against the Safety Assessment Principles (SAPs) and whether the proposed design reduces risks to As Low As Reasonably Practicable (ALARP).
- Reviewing details of the Hitachi-GE design controls and quality control arrangements to secure compliance with the design intent.
- Establishing whether the system performance, safety classification, and reliability requirements are substantiated by the detailed engineering design.
- Assessing arrangements for ensuring and assuring that safety claims and assumptions are realised in the final as-built design.
- Resolution of identified nuclear safety and security issues, or identifying paths for resolution.

5. This is my report from the ONR's Step 4 assessment of the Hitachi-GE UK ABWR design in the area of radiological protection.

6. All of the regulatory observations (ROs) issued to Hitachi-GE as part of my assessment are also published on our website, together with the corresponding Hitachi-GE resolution plan. There were no regulatory issues (RIs) submitted to Hitachi-GE from radiological protection, however, there were RIs which Hitachi-GE responded to and successfully closed which underpin the radiological protection assessment.

1.2 Scope

7. The scope of my assessment is detailed in my assessment plan (Ref. 3).

8. The Step 4 assessment assessed whether occupational and public exposures to ionising radiations are ALARP during normal operation. This assessment re-visited the Step 3 assessment in light of detailed evidence submitted by Hitachi-GE and assessed the robustness of that evidence for potential exposures. The assessment both developed areas identified during Step 3 and focused on areas not covered in Step 3, such as occupational exposure associated with the most dose intensive activities, fuel route, waste handling, shielding, ventilation, contamination control, radiological instrumentation, and decommissioning. Other assessors looked at accident risk and associated dose consequences in relation to the analysis of the Level 3 Probabilistic Safety Assessment (PSA) during Step 4.The analysis of on-site exposures from the Level 3 PSA assessment is reported in the Radiological Consequence Assessment Excluding Offsite Level 3 PSA (Ref.4) and the Level 3 PSA assessment for off-site consequences is reported in the GDA report for Step 4 assessment of Probabilistic Safety Assessment for the UK ABWR (Ref. 5).

9. There are matters relevant to radiological protection that cannot be adequately assessed during GDA, as they are directly related to the operating regimes selected by future licensees. This assessment has been primarily focused on the radiological risks associated with physical design features associated with the UK ABWR, rather than the specific working practices where there is an inherent radiological risk, because these practices will be subject to change based on licensee operating preferences. However, Hitachi-GE has submitted examples of specific working practices for some tasks in order to demonstrate that the magnitude of doses incurred by personnel align with relevant legislation and standards. This is to demonstrate the effectiveness of design features which are incorporated within the UK ABWR plant. These examples have been a useful factor in demonstrating the application of the ALARP principle.

10. The assessment was carried out in consultation with fellow assessors in ONR and the EA in other topic areas, such as:

- PSA.
- Deterministic Safety Analysis (fault studies).
- Reactor Chemistry.
- Radioactive Waste Management & Decommissioning.
- Mechanical Engineering.
- Human Factors.
- Environment.
- Control & Instrumentation (C&I).

11. A number of other topic areas in the SAPs (Ref. 6) have some relevance to radiological protection, such as safety cases, siting (not a direct issue for the GDA process), key principles, integrity of metal components and structures, layout, control of nuclear matter, control and instrumentation of safety-related systems, containment and ventilation, heat transport systems, radioactive waste management, and decommissioning. The lead for these topic areas was taken by ONR assessors in other disciplines and this assessment contributed to radiological protection aspects of these topic areas as appropriate.

12. Further to this my assessment included reviews of documentation provided by Hitachi-GE including relevant chapters of the PCSR, supported by Topic Reports and where appropriate Basis of Safety Case reports along with supporting references where it was felt appropriate to inform my assessment. As detailed above the areas reviewed for Step 4 included those identified at Step 3 and specifically:

- RO-ABWR-0014 "UK ABWR Radiological Safety Case: Project Plan and Delivery".
- RO-ABWR-0064 "Design approach to identification and provision of both permanent and temporary features necessary for the adequate control of radioactive contamination across the full lifetime of UK ABWR".
- RO-ABWR-0065 "Demonstration of adequate design and implementation of inherently safe techniques and structures to minimise radiation dose rates/short paths via through wall penetrations".

13. The areas identified above are referenced in the Step 4 Radiation Protection Assessment Plan (Ref. 3) and augment the general areas of assessment which include:

- Shielding Design.
- Designation of Radiological Areas and Zoning to ensure the design reduces exposures and therefore risk as low as reasonably practicable.
- Direct Radiation Exposure to the General Public from Operations.
- Exposures to the workers and others on site from both external and internal sources of radiation during routine operations.
- Post-Accident Accessibility design and provision to reduce risk as low as reasonably practicable to allow for actions to be carried out and allow plant to returned to a safe state.
- Review of Hitachi-GEs process for obtaining, collating, analysing and use of International Operational Experience (OPEX) in the design of UK ABWR.

14. Further to the areas listed above a number of developing areas have also been identified and these include:

- The inclusion of a Bottom Drain Line (BDL) within the UK ABWR design and its justification in relation to radiation exposure to workers during normal operations and also during potential fault scenarios.
- Heating Ventilation and Air Conditioning (HVAC) design and provisions for the UK ABWR buildings in relation to contamination control. This was primarily assessed through RO-ABWR-0064.

15. There are currently no Boiling Water Reactors (BWRs) in the UK. Due to the UKs limited experience of BWR technology, it should be acknowledged that the a large part of UK ABWR reactor island design is novel to the UK. As such the areas stated above and specifically the Requesting Parties responses to RO-ABWR-0064 and RO-ABWR-0065 are seen as both targeted and proportionate with respect to assessment of this design. Opportunities for further investigation of the safety case and supporting documentation have been undertaken, using interactions with other specialisms such as: Radioactive Waste & Decommissioning, Mechanical Engineering, Human Factors and C&I .

16. The scope of my assessment is appropriate for GDA because of the nature of the reactor design in relation to UK reactor experience. By focusing on the areas above and specifically the ROs generated in Step 3 I have been able to assess the contamination control aspects of the design, focusing on the hierarchy of control measures by: removal of the source of ionising radiation, reduction of the source, prevention of exposure, mitigation of exposure and control through administrative means or finally consideration of Personal Protective Equipment (PPE). In my opinion this is proportionate because it provides both a broad high level review and allowed for a number of deep slice reviews to take place.

17. I have also utilised a Technical Support Contractor (TSC) to carry out detailed review of Shielding design along with Direct Radiation Exposure to the General Public and a review of OPEX from European BWR experience.

18. Whilst the TSCs undertook detailed literature and technical reviews, these reviews were under close direction and supervision by ONR and the regulatory judgments on the adequacy, or otherwise, of the radiological protection aspects of the UK ABWR were made exclusively by ONR. The findings relating to radiological protection aspects of the literature and technical reviews by TSCs are incorporated into multiple sections of my report as appropriate.

19. Following due process, feedback on progress and outcomes of TSC work were provided to Hitachi-GE throughout the process.

1.3 Method

20. My assessment complies with internal guidance on the mechanics of assessment within ONR (Ref. 7).

2 ASSESSMENT STRATEGY

2.1 Standards and Criteria

21. The standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAPs) (Ref. 6), internal TAGs (Refs. 8, 9, 10, 11, 12 and 13), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites as well as international practices.

2.1.1 Safety Assessment Principles

22. The key SAPs applied within the assessment are included within annex 1.

2.1.2 Technical Assessment Guides

23. The Technical Assessment Guides (TAGs) that have been used as part of this assessment are set out in annex 2.

2.1.3 National and International Standards and Guidance

24. The international standards and guidance that have been used as part of this assessment are set out in annex 3.

25. The framework underpinning all of the standards and criteria above are the principles of radiological protection, namely, justification, optimisation and limitation.

- Exposures to radiation should be justified. Justification is not regulated by ONR and is not considered in the SAPs. Justification for electrical power generation is covered by the Department for Business, Energy & Industrial Strategy (BEIS).
- Exposures to ionising radiation must be optimised. Radiation exposures must be restricted “so far as is reasonably practicable” under The Ionising Radiations Regulations (IRR99) (Ref.14), that is, doses should be “as low as reasonably practicable” (ALARP). In this report the UK term “ALARP” is taken to be synonymous with the international term “ALARA” (“as low as reasonably achievable”) and with “SFAIRP ” (“so far as is reasonably practicable”).
- Exposures to ionising radiation must be limited so that they do not exceed the statutory dose limits in IRR99 (Ref. 14). Clearly this should not be an issue for a modern nuclear power plant under normal operation, as is indeed the case for the UK ABWR.

26. Radiological protection will make a contribution to fulfilling the expectations of some of the fundamental principles in the SAPs (Ref. 6), although radiological protection, or indeed any other single topic area, could not fulfil those expectations alone. The key fundamental principles that have some relevance to radiological protection are FP.3 to FP.8. The radiation protection principles RP.1 to RP.7 are for normal operation, accident conditions, designated areas, contaminated areas, decontamination, shielding and hierarchy of control measures; all of these areas were covered by the assessment. This section of the SAPs on Radiation Protection (Ref. 6) also refers to IRR99 (Ref. 14), and in particular to the Approved Code of Practice (ACOP) and guidance to IRR99 on the hierarchy of control measures (RP.7) in regulation 8 (Ref. 15). Criticality is not specifically covered in this assessment. It is covered in the Reactor Fuel & Core assessment (Ref. 16).

27. All the numerical targets and legal limits (NT.1 Targets 1 to 9 and NT.2) are relevant to a degree. The radiological protection assessment focused on NT.1 Targets 1 to 3 regarding impacts to people during normal operation, and NT.2 regarding time of exposure of employees in high dose rate locations. The lead for design basis fault

sequences and Level 3 Probabilistic Safety Assessment (PSA) was taken by another ONR assessor in cooperation with other disciplines. The radiological protection assessment contributed to NT.1 Target 4 regarding radiological consequence assessment of design basis fault sequences and to NT.1 Targets 5 to 9 regarding radiological consequence assessment of accidents (including Level 3 PSA), which is reported in the Step 4 reports for Radiological Consequence Assessment Excluding Offsite Level 3 PSA and Assessment of PSA for the UK ABWR, (Ref. 4, 5). These principles, targets and limits were assessed on a sampling basis to the extent that they could be accommodated within the GDA process. They will also need to be considered during the site specific phase.

28. In IRR99 (Ref. 14), the annual limit on effective dose for workers is 20 $mSvy^{-1}$. The Basic Safety Level (BSL) as specified in the SAPs is the level of dose above which the risk of harm is intolerable and for workers who are working with ionising radiation during normal operation (NT.1 Target 1), it is the same value as the annual dose limit under IRR99 (Ref. 14), namely 20 $mSvy^{-1}$. The BSL for groups of persons working with ionising radiation during normal operation is half of that value, namely, 10 $mSvy^{-1}$ (NT.1 Target 2). The BSL for other employees on-site during normal operation (e.g. workers not working with ionising radiation, other employees on the site) is 2 $mSvy^{-1}$ (NT.1 Target 1). The BSL for members of the public off the site during normal operation is the same as the public dose limit under IRR99 (Ref. 14), namely 1 $mSvy^{-1}$ (NT.1 Target 3).

29. The Basic Safety Objective (BSO), as specified in the SAPs, is the level below which it would not be reasonable use of ONR resources to seek further reductions in radiation doses from operators. Nevertheless, the principle of ALARP still applies to operators at levels below the BSO which may drive doses down below the BSO. The BSO for workers who are working with ionising radiation during normal operation is one twentieth of the BSL / annual dose limit under IRR99 (Ref. 14), namely 1 $mSvy^{-1}$ (NT.1 Target 1). The BSO for groups of persons working with ionising radiation during normal operation is also one twentieth of the BSL, namely, 0.5 $mSvy^{-1}$ (NT.1 Target 2). The BSL for other persons on-site during normal operation (e.g. workers not working with ionising radiation) is again one twentieth of the BSL, namely 0.1 $mSvy^{-1}$ (NT.1 Target 1). The BSO for members of the public off the site during normal operation is more challenging in that it is a much lower proportion (one fiftieth) of the BSL / public dose limit under IRR99 (Ref. 14), namely 0.02 $mSvy^{-1}$ (NT.1 Target 3).

30. BSLs for design basis fault sequences (NT.1 Target 4) for any people on or off the site are expressed in terms of radiation dose and are dependent on frequencies of initiating fault sequences. However, there is only one BSO for people on the site, and a different one for people off the site (also expressed in terms of radiation dose), and these are independent of frequencies of initiating fault sequences. BSLs and BSOs for accident conditions for any people on the site or any people off the site (NT.1 Targets 5, 6,7 and 8 respectively) are dependent on frequencies of accidents.

31. The dose criteria for the BSLs and BSOs encompass both external and internal doses, although clearly the shielding assessment only considered exposure to external radiation.

32. The TAGs of most relevance to the assessment are on fundamental principles (Ref. 8), demonstration of ALARP (Ref. 9), radiological protection (Ref. 10), radiation shielding (Ref. 11), radiological analysis during normal operation (Ref. 12), and radiological analysis during fault conditions (Ref. 13).

33. The relevant fundamental principles, radiation protection principles, criticality safety principles and numerical targets and legal limits from the SAPs (Ref. 6) are summarised in Annex 1. Relevant Western European Nuclear Regulators' Association (WENRA) and International Atomic Energy Agency (IAEA) and the Organisation for

Economic Co-operation and Development (OECD) references are in Annex 3. The table stipulated within the step 4 plan (Ref. 3) also indicates the contributions made by these principles, targets and limits to the Step 4 radiological protection assessment.

34. The principal standards and criteria for judging whether ALARP has been met are the ACOP and guidance to IRR99 (Ref. 15), supplemented by additional guidance on ONRs website (including the TAGs). In addition, IRR99 (Ref. 14) require a hierarchical approach to control exposure: first, exposures should be restricted by engineered controls and design features (and in addition, by the provision and use of safety features and warning devices); secondly, by supporting systems of work; and thirdly and lastly, by the provision of personal protective equipment.

35. The principal standards and criteria for judging whether ALARP has been met for intervention personnel during accident conditions is in the Radiation (Emergency Preparedness and Public Information) Regulations 2001 (Ref. 17), supplemented by additional guidance on HSE's website (Ref. 18). This is further supported by Provisional HSE Internal Guidance on Dose Levels for Emergencies, HSE 2008 (Ref. 19) and guidance published by Public Health England's Centre for Radiation, Chemical and Environmental Hazards (PHE-CRCE) on controlling doses for people on-site during radiation accidents (Ref. 20).

36. When judging against the ALARP principle, caution should be used to distinguish between dose and risk. The general duties of employers to their employees and other persons in Sections 2 and 3 respectively of the Health and Safety at Work etc. Act 1974, as amended (Ref. 21), refer to risks as do the expectations in many of the SAPs (Ref. 6). However, the duties of radiation employers in IRR99 (Ref. 14) and standards in some of the SAPs (Ref. 6) refer to radiation exposures and not just to the implied risk. The hierarchy of control measures in IRR99 (Ref. 14) is also applicable here, as the Approved Code of Practice (ACOP) to regulation 8 advises radiation employers to give priority to improving engineering controls and adopting other means of restricting exposure over and above dose sharing between employees (Ref. 15). If a choice has to be made between restricting exposures to individuals or to groups of employees then priority should always be given to restricting exposures to individuals. In contrast to this, under accident conditions, the risk is determined by both the magnitude of the dose and the probability of its occurrence. For the purposes of ALARP, the risk of harm to an individual from whole-body exposure is taken to be directly proportional to that dose.

37. The ALARP principle applies to the exposure of members of the public. The regulation of public radiation exposure during normal reactor operation is shared between the EA and ONR, where IRR99 (Ref. 14) is enforced by ONR, and EPR16 (Ref. 22) is enforced by the EA. IRR99 (Ref. 14) require dose constraints to restrict exposure to ionising radiation at the planning stage where it is appropriate to do so. The guidance to IRR99 (Ref. 15) advises that a constraint for a single new source should not exceed 0.3 mSv per year for members of the public. This is reinforced in the SAPs (Ref. 6) in relation to NT.1 Target 3 and advises that ONRs view is that a single source should be interpreted as a site under a single duty holder's control, since this is an entity for which radiological protection can be optimised as a whole. However, Public Health England's Centre for Radiation, Chemical and Environmental Hazards (PHE-CRCE) has recommended that the dose constraint for members of the public from new nuclear power plants (NPPs) should be 0.15 mSv per year (Ref. 23).

38. The ALARP principle also applies to manufacturers, etc. Section 6 of HSWA74 (Ref. 21) places general duties on manufacturers, etc. as regards articles and substances for use at work and duties on any person who designs, manufactures, imports or supplies any article for use at work. Where that work is with ionising radiation, the duty is modified to apply to articles for use at work by IRR99, regulation 31 (Ref. 14). This requires manufacturers, etc. to apply the ALARP principle in that there is a duty to

ensure that any such article is so designed and constructed as to restrict so far as is reasonably practicable, the extent to which employees and other persons, are or are likely to be, exposed to ionising radiation. Therefore, the requirement in law to keep radiation exposures ALARP applies not only to the licensee of a NPP, but also to the designer of that NPP.

2.2 Use of Technical Support Contractors (TSCs)

39. It is usual in GDA for ONR to use TSCs. For example TSCs can provide additional capacity, provide access to independent advice and experience, as well as specialist analysis techniques and models, and enable ONR's inspectors to focus on regulatory decision making etc.

40. Table 1 presents the broad areas for which technical support was used. Nuclear Technologies Technischer Überwachungsverein Süddeutschland (TÜV SÜD) was chosen through competitive tender to support ONR in its assessment of the radiological safety aspects of the UK ABWR design. The support specifically focused on:

- Shielding design and assessment
- Designation of radiological areas and associated zoning
- International OPEX with particular focus on German and Swedish experience in relation to radiological protection
- Direct dose to the public from the operational facility

41. Nuclear Technologies TÜV SÜD augmented the knowledge and experience of ONR inspectors bringing detailed knowledge of shielding codes and radiological modelling capability and in doing so also provided additional capacity during the period of assessment. The parent company TÜV SÜD review technical and operational support and advice to the German BWR fleet currently undergoing Post-Operative Clean Out (POCO) and decommissioning. This operational experience was valuable to ONR as current UK knowledge and experience of BWR design is limited.

Description of work - Key Deliverable	TSC	Start Date	Completion Date
Assessment of Reports on Shielding Specification and Design leading to recommendations with regard to acceptability	Nuclear Technologies (TÜV SÜD)	June 2016	July 2017
Assessment of Reports on Radiological Zoning leading to recommendations with regard to acceptability	Nuclear Technologies (TÜV SÜD)	June 2016	July 2017
Assessment of Reports on Public Exposure to Direct Shine from Normal Operations	Nuclear Technologies (TÜV SÜD)	June 2016	July 2017
Assessment of International and German OPEX in relation to BWRs	Nuclear Technologies (TÜV SÜD)	June 2016	July 2017

Table 1: TSC work packages in support of ONR Radiological Protection assessment of the UK ABWR.

2.3 Integration with Other Assessment Topics

42. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot therefore be carried out in isolation as there are often safety issues of a multi-topic or cross-cutting nature. The following cross-cutting issues have been considered within this assessment:

- Source Terms - working with Reactor Chemistry and Structural Integrity.
- HVAC - working with Mechanical Engineering.
- Bottom Drain Line (BDL) - working with Reactor Chemistry, Structural Integrity and Nuclear Liabilities Regulation (NLR).
- Fine Motion Control Rod Drive (FMCRD) - working with Mechanical Engineering, NLR, Conventional Safety, Human Factors and the EA.
- Materials Selection - working with Reactor Chemistry and Structural Integrity.
- Radioactive Waste - working with NLR, Reactor Chemistry and the EA.
- Decommissioning – working with NLR, Reactor Chemistry, Human Factors, Conventional Safety and the EA.
- Emergency Evacuation times – working with Human Factors.
- Post-Accident Analysis – working with Fault Studies and PSA.

2.4 Sampling Strategy

43. It is seldom possible, or necessary, to assess a safety case in its entirety, therefore sampling is used to limit the areas scrutinised, and to improve the overall efficiency of the assessment process. Sampling is done in a focused, targeted and structured manner with a view to revealing any topic-specific or generic weaknesses in the safety case.

44. The sampling strategy for this assessment was to utilise the findings from Step 3 and whilst using the original broad areas identified in Section 1.2 of the Introduction use these areas as the primary focus for deep slice reviews. Although the Requesting Party identified a number of areas to present evidence in relation to RO-ABWR-0064 and RO-ABWR-0065, ONR reserved the right to request evidence for additional areas seen as appropriate and proportionate, to ensure a meaningful assessment. I also ensured close interactions with fellow inspectors within the specialisms identified in my Step 4 plan to provide assurance in the wider Safety Case context.

2.5 Out of Scope Items

45. Table 2 sets out the items have been agreed with Hitachi-GE as being outside the scope of GDA.

Out of Scope items/areas	Summary of rational
Assessment of Internal doses to the public under normal operation	This is out of vires for ONR and is covered under NRW with support from the EA through the Generic Environmental Permitting process.
Environmental and Security aspects of the UK ABWR design	The environmental aspects are covered under the vires of NRW with support from the EA through the Generic Environmental Permitting process. The Security aspects are covered within the ONR Security Step 4 Assessment Report
Criticality Safety Assessment	Criticality Safety Assessment has been covered within the Fuel & Core Assessment Report (Ref 16).

Table 2: Out of scope items

3 REQUESTING PARTY'S SAFETY CASE

46. The Requesting Party's (RP's) safety case for radiological protection, which was assessed for Step 4, is documented in Chapter 20 (entitled Radiation Protection) of the PCSR, Rev B consisted of 7 documents each presenting a separate sub-chapter (Ref 24, 25, 26, 27, 28, 29 and 30). This version of the PCSR (Ref 24, 25, 26, 27, 28, 29 and 30) did not contain sufficient information upon which an adequate assessment of the design could be undertaken. As a result additional information was obtained through assessment of supporting documentation including Topic Reports and where required through the issue of RQs and ROs. The PCSR Rev C (Ref. 31) was updated in September 2017 and provided a consolidation of material generated in response to these RQs and ROs. My assessment is based on PCSR Rev C the supporting references and topic reports along with responses to RQs and ROs.

47. Within PCSR Chapter 20 Rev C, there are 13 sub-chapters. Sub-chapters 20.2 to 20.11 deal with the following specific areas of radiological protection which shall be discussed in greater detail during this chapter. Sub Chapter 20.1, 20.12 and 20.13 are the introduction, conclusion and references, respectively.

- Purpose and Scope.
- Definition of Radioactive Sources.
- Strategy to ensure that the Exposure is ALARP.
- Protection and Provisions against Direct Radiation and Contamination.
- Radiation and Contamination Monitoring of Occupational Exposure.
- Dose Assessment for the Public from Direct Radiation.
- Worker Dose Assessment.
- Post-Accident Accessibility.
- Assumptions, Limits and Conditions for Operation.
- Summary of ALARP Justification.

48. There are three appendices for Chapter 20 which provide information regarding the document map for radiation protection safety reports, key links with other PCSR chapters as well as representative safety functional claims in relation to radiation protection.

3.1 Purpose and Scope

49. In sub-Chapter 20.2 (Ref. 31), Hitachi-GE outlines the specific purpose of this chapter with regard to the UK ABWR safety case and provides the four high level claims for radiation protection.

- RP-C1: External and internal doses to workers are ALARP and meet the regulatory requirements during normal operation.
- RP-C2: External doses to the public are ALARP and meet the regulatory requirements during normal operation.
- RP-C3: External and internal doses to workers are ALARP and meet the regulatory requirements during design basis faults, beyond design basis faults and severe accidents.
- RP-C4: External and internal doses to the public are ALARP and meet the regulatory requirements during design basis faults, beyond design basis faults and severe accidents.

50. An outline of Chapter 20 is also provided within this sub-chapter.

3.2 Definition of Radioactive Sources

51. In sub-chapter 20.3 (Ref. 31) Hitachi-GE provides relevant overview of the UK ABWR design source terms. Due to the complexity there are four design source terms.

- Primary Source Term (PST) – The level of mobile activity within the nuclear boiler system; this source term covers the reactor water and steam.
- Process Source Term (PrST) – This determines the concentration of various nuclides within the UK ABWR plant system.
- Deposit Source Term (DST) – The source term reviews the concentration of nuclides that accumulate on both the internal pipework surfaces within various systems and the fuel cladding.
- End User Source Term (EUST) - Defined as the final level of activity considered for a particular assessment within a technical area of the safety and environmental case for the UK ABWR.

52. For each of the above design source terms there are two levels of source term values defined.

- Best Estimate (BE) - Source Term that is expected to be observed during the normal operation of the UK ABWR.
- Design Basis (DB) - Source Term that gives a conservative maximum value which can be considered to be a bounding limit for the plant design.

BE and DB source terms can then be derived and used as appropriate for the following operational phases.

- Start Up (SU) – The transition from Outage to the Power Operation.
- Power Operation (PO) – The reactor operating at a steady power.
- Shut Down (SD) – The transition from Power Operation to Outage.
- Outage (OT) – The phase where the reactor has been shut down for refuelling and maintenance.

53. Cycle Average (CA) - Average Source Term activity observed over the duration of a full 18 month cycle (composed of all the above phases).The EUST has been produced for specific uses within the PCSR, such as Radiological Protection EUST, Decommissioning EUST, etc. A specific document detailing the EUST for Radiological Protection (Ref. 32) is discussed within the sub-chapter. The document is derived from the PST and the DST radionuclide concentrations that are relevant to the piping and/or equipment locations of interest. Appropriate radionuclides and radioactive concentrations are selected to aid in dose assessments relating to:

- Radiological Shielding.
- Radiation and Contamination Zoning.
- Worker Dose Assessment (Internal and External).
- Public Dose assessment (External).

54. A summary of the general considerations of the EUST for radiological protection for each system (water, steam, off-gas, HVAC, liquid & solid waste management and deposition) is also provided.

3.3 Strategy to Ensure that the Exposure is ALARP

55. In sub-chapter 20.4 (Ref. 31) Hitachi-GE outlines the relevant regulatory requirements for the UK ABWR; in terms of radiological protection:

- International recommendations (ICRP 2007 Publication 103 (Ref. 33)).
- European requirements (Council Directive 2013/59/Euratom (Ref. 34)).
- UK Legislation (Ionising Radiations Regulations 1999 (Ref. 14)).

56. The chapter goes on to detail the ICRP principle of radiation protection (justification, optimisation and limitation).

57. In sub-chapter 20.4 (Ref. 31) Hitachi-GE outlines its ALARP strategy for employers working with ionising radiation, other employees on the site and the public. For each case the strategy is split into four stages.

- Reference ABWR Plant – A plant design has been used as a starting point for the UK ABWR design (this is Kashiwazaki Kariwa (KK) Unit 6 and 7 in majority of cases (see section 4.2.7.2 for further information on choosing these reactors)).
- Design Criteria and Good Practice – This incorporates relevant good practice (from Japanese and UK nuclear industry) to be used effectively within the design of the UK ABWR (i.e. ERIC-PD tool).
- Implementation – Once mitigating options have been identified and incorporated within the design of the UK ABWR the collective dose is calculated.
- Demonstration of ALARP – A further review is undertaken to see if additional risk reduction measures are applicable. Only when no further improvements in dose reduction are reasonably practicable is the UK ABWR design considered ALARP.

3.4 Protection and Provisions against Direct Radiation and Contamination

58. In sub-chapter 20.5 (Ref. 31) Hitachi-GE outlines the hierarchy of control measures the UK ABWR design shall incorporate.

59. This section summarises the specific mitigating options (design engineered features and administrative controls) that minimise direct radiation and containment for specific radioactive and contaminated components.

60. Appropriate examples are provided of mitigating options that shall be used within the UK ABWR design.

61. In sub-chapter 20.5 (Ref. 31) Hitachi-GE provides information on the radiological and contamination zoning for the UK ABWR; the relevant regulatory requirements relate to the Ionising Radiations Regulations 1999 (Ref. 14).

62. Hitachi-GE also provided information on the five types of radiation shielding within the UK ABWR.

- Reactor Shielding Wall.
- Primary Shielding.
- Secondary Shielding.
- Auxiliary Shielding.
- Shielding by Water.

63. Radiation shielding thicknesses have been calculated using Monte-Carlo N-Particle transport code (MCNP 5) which is an appropriate industry standard computer code.

3.5 Radiation and Contamination Monitoring of Occupational Exposure

64. In sub-chapter 20.6 (Ref. 31) Hitachi-GE provides information on the approach to be undertaken for monitoring of the UK ABWR for radiation and contamination.

65. Hitachi-GE outlined the relevant regulatory requirements for the UK ABWR; in terms of monitoring of occupational exposure:

- Ionising Radiations Regulations 1999 (Ref. 14).
- IAEA International Basic Safety Standards (Ref. 35).
- IEC Standards (Ref. 36).

66. Hitachi-GE also provided details of the design strategy for monitoring of occupational exposure, claiming that the UK ABWR will be capable of providing measurements of the following radiological sources.

- Direct radiation.
- Airborne contamination.
- Surface contamination.

67. Hitachi-GE claims that monitoring of the above radiological sources shall be achieved through the use of both portable and installed monitoring equipment. Two methodologies shall work in unison:

- Trend Monitoring – Information is trended and recorded over time to help instigate if further actions need to be undertaken if there is a change in radiation and contamination levels. This is usually achieved through installed monitors.
- Accurate Measurement – Precise values of radiation and contamination levels are measured to help with work planning. This is usually achieved through portable monitors.

3.6 Dose Assessment for the Public from Direct Radiation

68. In sub-chapter 20.7 (Ref. 31) Hitachi-GE provides an overview of the public dose assessment from direct radiation of the UK ABWR design. The PCSR specifically uses examples for evidence in support of its assertions such as the following:

- Reactor building.
- Turbine building.
- Radwaste building.
- Condensate Storage Tank (CST).
- Suppression Pool water surge Tank (SPT).

69. Sub-chapter 20.7 (Ref. 31) states that mitigating options have been undertaken to reduce dose to the public.

3.7 Worker Dose Assessment

70. In sub-chapter 20.8 (Ref. 31) Hitachi-GE provides information on worker dose assessment for the UK ABWR within the following areas:

- Employees working with ionising radiation.
- Other employees on site.

71. The UK ABWR worker dose assessment provides evidence that:

- Radiological Protection is optimised such that doses are ALARP.
- Dose uptake to the most exposed group is minimised.

72. When calculating the reference dose for the UK ABWR, the OPEX information used was from the reference plant KK-7 (see section 4.2.7.2 for further information). The highest eight worker activities during an outage phase were used as representative examples to demonstrate that the worker dose is ALARP.

73. In sub-chapter 20.8 (Ref. 31) an example of the highest worker dose activity is provided (Reactor opening / series work) to demonstrate that the worker dose is ALARP.

74. For calculating dose to other employees on site from direct radiation a computer code is applied based on the reference design and UK ABWR specific information. An ALARP review is also undertaken to demonstrate that no further risk reduction is achievable.

3.8 Post-Accident Accessibility

75. In sub-chapter 20.9 (Ref. 31) Hitachi-GE identifies representative DBA and SA sequences that would necessitate direct intervention by personnel and the SSCs used to mitigate the consequences.

76. In sub-chapter 20.9 (Ref. 31) relevant regulatory requirements for the UK ABWR are outlined; in terms of radiological emergency these are:

- REPPIR 2001 (Ref. 17).
- IRR 1999 (Ref. 14).

77. Hitachi-GE provided information on access arrangements as well as examples for both DBA and SA sequences.

3.9 Assumptions, Limits and Conditions for Operation

78. In sub-chapter 20.10 (Ref. 31) Hitachi-GE identifies the assumptions, limits and conditions for operations from a radiation protection perspective for the UK ABWR GDA design.

79. Criteria set out in IRR 1999 (Ref. 14) have provided the limits and constraints regarding dose for the UK ABWR.

80. Hitachi-GE stated assumptions that have been made for the radiological protection design of the UK ABWR.

3.10 Summary of ALARP Justification

81. In sub-chapter 20.11 (Ref. 31) Hitachi-GE provides a high level overview of how radiological protection has complied with the ALARP principle.

A high level review of the UK ABWR ALARP process is provided. A list of UK ABWR improvements has been supplied to corroborate the argument.

4 ONR STEP 4 ASSESSMENT

82. This assessment has been carried out in accordance with ONR internal guidance on the “Purpose and Scope of Permissioning” (Ref. 37).

4.1 Scope of Assessment Undertaken

83. As stated earlier, the scope of the assessment is in line with my Step 4 Radiological Assessment Plan (Ref. 3).

4.2 Assessment

84. My Step 4 Radiological Protection Assessment Plan (Ref. 3) identified a number of topics for assessment. Understanding of the hazard begins with the source term, which in turn allows for consideration of the: the designation of areas (zoning classification); the design and provision of shielding; optimisation of radiation exposure (including the specific design for contamination control and prevention of access to high dose and dose rate areas); the generic design of HVAC; and post-accident accessibility.

85. Prior to commencement of the assessment I identified the need for additional technical support from TSCs in order to carry out detailed assessment of shielding, direct dose to the public, zoning of radiological areas and provide a general OPEX review. Plume dispersion modelling and Level 3 PSA consequence assessment is captured in the Step 4 PSA report (Ref. 5) and the Radiological Consequence Assessment Excluding Offsite Level 3 PSA (Ref. 4); I refer to both reports. Reports provided by Hitachi-GE covering these topics have been subject to detailed review and analysis by TSCs and assessment by the Radiological Consequences assessor within ONR. My assessment of radiological protection has been augmented by their analysis.

86. In order to obtain further information to support the claims outlined in the PCSR, I raised a number of RQs and ROs, both in a leading role and sometimes as a supporting assessor. The information provided by Hitachi-GE in response to these RQs and ROs alongside the PCSR Rev B and updated PCSR Rev C, Topic Reports, Basis of Safety Case, and other supporting references constituted the arguments and evidence which have been used in my assessment.

4.2.1 Normal Operation – Radiation Sources

87. Management of radiation sources associated with the operation of a nuclear reactor is a fundamental aspect of radiological protection at NPPs. As it is not practicable to eliminate the sources of ionising radiation at NPPs, the emphasis must be on reducing the magnitude of radiation sources in order to reduce radiation levels and therefore minimise exposures of personnel and the public to ionising radiation. Although many measures, which can be taken to reduce radioactive sources associated with an ABWR reactor, are related to the operating regime which is selected for the plant and so depend on the decisions taken by future licensees, there are aspects that are related to the physical design itself and these have been the subject of my assessment.

88. My assessment in this area is structured in two parts: the management of the source term information and the measures in place to reduce the radiation source term associated with the generic plant.

4.2.1.1 Assessment - Information on the Source Term

89. The PCSR Chapter 20.3 Rev C (Ref. 31) provides a summary of the radiation sources associated with normal operation throughout commissioning, operation, maintenance, outage for refuelling and decommissioning. It also provides a summary of the source terms' radiological significance relevant to radiological protection and a summary of measures which restrict the exposure of workers and the public. It segregates the potential effects on each of the exposed groups (i.e. workers and the public). A definitive list of source terms used for radiological protection for shielding and dose assessments in support of the UK ABWR design are presented in the EUST for Radiation Protection (Ref. 32).

90. Additional information on the source terms had to be obtained from Hitachi-GE during Step 3 and into Step 4 through the drafting of an RI by the ONR Reactor Chemistry assessors. This required Hitachi-GE to make significant refinements in relation to the structure and presentation of source term information. This is reported in detail within the Reactor Chemistry Step 4 report (Ref. 38).

91. As Hitachi-GE developed a revised source term strategy and associated documentation, I along with ONR and EA colleagues have sought clarification regarding a number of points associated with specific plant areas, SSCs and Hitachi-GE technical assessments. This was done by raising several RQs including RQ-ABWR-0722 (Ref. 39). Information obtained from this RQ and others was used in the reviews of shielding, direct dose to the public, radiological zoning and dose assessment carried out by the TSC and ONR. The public dose and shielding assessments both constituted sampling by 'deep slice review'.

92. Topics covered by RQs included: clarification of the effectiveness of the reactor water chemistry regime, application of source terms in identifying contamination and radiation zoning limits, and application of the provenance of source terms in shielding assessments.

93. It should be noted the source term reports do not cover those sources of radiation in the form of sealed/special form sources installed within the reactor such as neutron sources used for flux stabilisation, or other sources used for radiography or instrument calibration and/or check functions. These were covered through the issue of RQ-ABWR-0612 and Hitachi-GEs subsequent response "Neutron Sources and other Radioactive Sources (Response to RQ-ABWR-0612)" (Refs. 40 and 41).

94. The UK ABWR is designed to operate under hydrogen water chemistry (HWC) to create a reducing low oxygen environment and therefore control trans-granular stress corrosion cracking (TGSCC) which has been identified as an issue in reactors using carbon steel and steel alloys. The implementation of HWC increases ^{16}N concentrations in the reactor primary coolant. ^{16}N is a relatively high energy gamma emitting nuclide and is therefore significant from a radiological protection perspective as this can have an impact on doses during operations.

95. Under the reducing environment nitrogen shifts from nitrate, which is non-volatile, to more volatile forms such as nitrogen oxides and ammonia. This increases the proportion of ^{16}N which moves into the steam phase. Following the use of HWC Noble Metal, specifically On-line NobleChem™ (OLNC) is to be applied which has the effect of reducing the amount of hydrogen required and thus lowering the level of ^{16}N carry over into the steam phase.

96. Along with ^{16}N, ^{60}Co is a major source of radiation exposure in all light water reactors. Zinc (Zn) addition is employed as an option to reduce surface ^{60}Co levels. High ^{60}Co concentrations in BWR water results in higher ^{60}Co uptake in surface corrosion films and hence higher radiation dose levels. The application of depleted Zn aids in reducing

radiation fields due to ^{60}Co uptake on surface films as Zn is preferentially incorporated into the corrosion films.

97. Source term definition is a formative step in understanding and deriving the safety requirements of any nuclear activity as it provides an understanding of the hazard and its magnitude, the basis of any risk assessment. In the PCSR (Ref. 31) the source term or radioactive inventories are used in a number of different assessment areas and radioactive inventories may be adapted to address specific purposes.

98. ONR and the EA routinely discussed the use of source terms during the assessments, particularly their application to public exposure from direct radiation and in relation to the EA's assessment of dose from discharges.

99. Different source terms have been derived differently depending on the nature of the source term. This may be from OPEX or through the application of computer codes as described later in this section. The development of the source term is detailed in a suite of tiered documents shown in Figure 1 (Ref. 42).

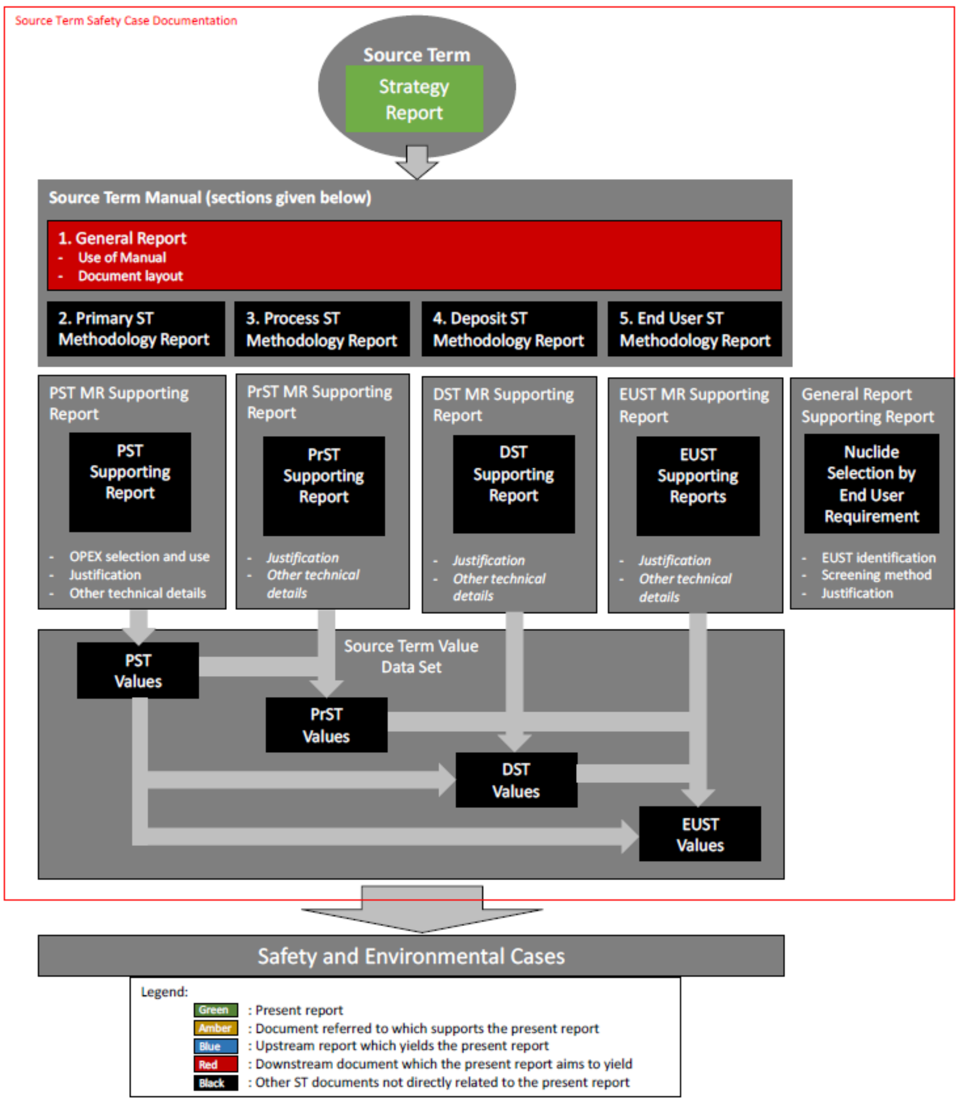

Figure 1: Overview of Hitachi-GE tiered Source Term document structure (Ref. 42)

100. Hitachi-GE defines a number of different values for each source term component within the Source Term Manual General Report (Ref. 43) as part of its suite of documents. The structure of the components is to allow for a range of values to be provided dependant on the nature of use. These Source Term Components are :

- Primary Source Term (PST) – Defined as the level of activity at outlets of the Reactor Pressure Vessel (RPV). The PST quantifies the concentration of each radionuclide present in the reactor water and reactor steam.
- Process Source Term (PrST) – Defined as the level of activity within each of the systems in the UK ABWR. The PrST quantifies the concentration of each radionuclide present within circuit pipes, ancillary equipment and plant systems.
- Deposit Source Term (DST) – Defined as the level of activity deposited within each of the systems in the UK ABWR. The DST quantifies the concentration of each deposited radionuclide on internal pipework, ancillary equipment, plant systems and fuel pins.
- End User Source Term (EUST) – Defined as the final level of radioactivity considered for a particular assessment within a technical area of the safety and environmental case for the UK ABWR.

101. For each of these source terms Best Estimate (BE) and Design Basis (DB) values were calculated. These are defined as follows:

- BE – Defined as the radionuclide concentration (in reactor water and/or steam) that is expected to be observed during the normal operation of the UK ABWR. Hitachi-GE present the BE Source Term as a realistic and reasonable value so as not to result in over specification of the plant systems. For example, the BE value for the PST is derived using a statistical analysis of relevant OPEX data, augmented with model calculations. While the specifics of the statistical analysis applied varies between key radionuclide groups, and between operating phases, generally an average value is determined for each plant in the OPEX inventory and then an overall average value for all the plants considered is determined.
- DB – The DB value for the PST is defined for each significant radionuclide and is the concentration of that nuclide (in reactor water and/or reactor steam) that gives a conservative maximum value which can be considered a bounding limit for the plant design. The value is derived using the same relevant OPEX with further statistical analysis. Hitachi-GE applies an adjustment for uncertainty and fuel failure. This is assessed in more detail within the Fuel & Core and Reactor Chemistry Reports (Refs 16 and 38). Hitachi-GE normalises these values against a design basis limit which is linked to noble gas inventory released from the plant following fuel failure.
- As described in paragraph 52, BE and DB values can then be applied to source terms for systems and components during the SU, PO, SD and OT phases of reactor operation. In addition, the CA Source Term is defined which is the average of the phases observed over the duration of a full cycle, i.e. approximately 17 months PO and 1 month SD/OT/SU.
- In addition, three categories of radionuclides are defined for each source term type: Corrosion Products (CP), Activation Products (AP) and Fission Products / Actinides (FP/ActP).

102. Where radionuclide specific OPEX is unavailable the ORIGEN code is used to determine a relationship between the radionuclide concerned and ^{60}Co, which Hitachi-GE has chosen as a radionuclide with high radiological significance.

103. The Radiation Protection EUST values are derived from the PrST and DST radionuclide concentrations that are relevant to piping and/or equipment locations of interest. For example, to derive shielding requirements for mobile radionuclides in the Main Steam System, the DB value for the PrST is used. For each radionuclide

category, the worst case concentration is assumed. For APs this is during PO, for CPs this occurs during SD and for FP/ActPs it may occur during PO or SD depending on the radionuclide. To assess shielding requirements due to accumulated activity in the Radwaste System, the DB value of the PrST is also used, but in this instance the CA value for all radionuclide categories is used. The justification for this is that the activity in these components accumulates through the SU, PO, SD and OT phases and that as these components are not connected to the primary system, they are not directly exposed to transient increases in activity.

104. The competent development and application of source terms is a key consideration for radiological protection at the design stage of nuclear facilities as they will form the design basis for shielding structures to protect personnel and the public. As part of a sample on this topic, Nuclear Technologies conducted a review of the source terms in support of an assessment of shielding provisions for the UK ABWR (Ref. 44) and judged that the calculations used to generate the source term have been undertaken using acceptable codes, methods and gamma spectra, when compared against relevant good practice in the UK. In the context of shielding design, the TSC specifically concluded the following:

- The process of interpreting OPEX data from the operational facilities and the application of that data within the context of the UK ABWR is appropriate.
- Where OPEX data might not be considered appropriate modelling has been used to generate appropriate data.
- All major contributions to the source term are appropriately accounted for, e.g. crud-burst on shutdown and start-up, fuel failures, presence of tramp uranium and variations in coolant chemistry.
- The lists of radionuclides contain all of the expected radionuclides with respect to external radiation.
- No concerns were raised during the review of the source terms used in defining the shielding provisions for the UK ABWR.
- Source terms are well defined and as a result of key assumptions it is apparent that they will be conservative when compared with more realistic source terms based upon observations on existing plants.
- It has confidence that the shielding provisions and the predicted dose rates for any given area of the plant will also be conservative with respect to protection of the public and personnel from external radiation.
- The transformation from radionuclide fingerprints to gamma emissions spectra for use in shielding calculations has also been considered in detail.

105. The TSC and ONR are generally satisfied with Hitachi-GEs evidence provided in these areas, however, a number of points are identified which will require consideration within the site licensing phase. These aspects included:

- Unlike the PrST, the DST for the UK ABWR is not directly derived from the PST. Instead, the DST is derived mostly from US BWR OPEX and where data is not available by calculation (using Studsvik BWRCrud, (Ref. 44)). In an operating reactor, the activity within the PST and the PrST define the DST. There is therefore a potential for misalignment in the source term derivations for the UK ABWR when considered as a whole. This will result in a difference in nuclide concentrations and proportions. This is not seen as significant due to the conservative nature of the derivation methods.

106. Further to the point identified immediately above, further proposals in the commissioning and operation of the UK ABWR made by Hitachi-GE will further reduce the deposit source term, including those nuclides which OPEX data has not been corrected or modified for application to the UK ABWR and this gives further confidence the values presented will be suitably conservative. These include:

- The use of hot functional testing using Depleted Zinc Oxide (DZO) to develop a zinc oxide layer prior to fuel being inserted into the reactor. The circuit should therefore be less prone to absorbed radioactivity during operations.
- The use of Stellite alternatives for hard-facing materials (e.g. for valve facings) to reduce the amount of cobalt in contact with the cooling water and reduce the amount of cobalt within the cooling water arising from erosion/corrosion that could potentially be activated when passing through the core.

107. Section 4.2.4 of this report provides further details of the assessment of the shielding source term, specifically with respect to the shielding assessment.

108. Furthermore, Hitachi-GE has produced documentation to provide evidence on how it manages and controls the defined source terms. These are detailed in “The Management of Source Terms” (Ref. 45).

109. I have assessed Hitachi-GE Source Term documentation against the ONR Standards on Radiation Shielding (Ref. 11) and considered aspects of the characterisation, suitability and conservatism, taking into account sensitivity and cliff edge effects.

110. In conclusion I am broadly satisfied with Hitachi-GEs approach to Source Term derivation and management and specifically with respect to Radiation Protection EUST as derived in relation to the GDA for UK ABWR.

4.2.2 Normal Operation – Demonstration that Worker Dose is ALARP

111. This part of my assessment focussed on the following aspects of the PCSR:

- Hitachi-GE’s approach to ALARP.
- Application of ALARP to reactor system design.
- Prioritisation of ALARP for work activities involving the highest doses.

112. The assessment was principally against the following standards:

- The Ionising Radiations Regulations (IRR99) (Ref.14) and Approved Code of Practice (ACOP) and guidance (Ref. 15), especially regulation 8.
- ONR’s Safety Assessment Principles (SAPs): SAPs Fundamental Principles FP.3, FP.4, FP.8 and Radiation Protection RP.1, RP. 7. (Ref. 6).
- NS-TAST-GD-005 (Rev 8) – ONR Guidance on the demonstration of ALARP (Ref. 9).
- NS-TAST-GD-038 (Rev 6) – Radiological protection (Ref. 10).
- NS-TAST-GD-043 (Rev 3) – Radiological analysis normal operation (Ref. 12).

113. It should be noted that the extremely significant contributions of source term minimisation and radiation shielding to delivering ALARP worker doses are covered in sections 4.2.1 and 4.2.4 respectively. Source term minimisation includes consideration of the water chemistry regime and material selection considerations. This section therefore deals mainly with analysis of the Hitachi-GE approach to ALARP. Some examples from the assessment are given. Where concerns on the approach to ALARP have contributed to ROs being raised, the relevant section is referenced providing further detail.

114. The application of ALARP was considered on both a system and operation basis. A number of systems and operations were then considered as a sampling to determine whether the design was ALARP from the point of view of worker dose.

115. Systems considered included the following:

- Solid waste management system.
- Liquid waste management system.

- Reactor Area (R/A), Turbine Building (T/B), Radioactive Waste Building (Rw/B) and Service Building (S/B) HVAC Systems.
- RPV BDL.

116. I undertook a review of the Topic Report: Demonstration to Ensure that External and Internal Doses are ALARP for all Relevant Buildings during System Start-up, Power Operation, Normal Hot Stand-by, System Shutdown and Outages excluding ILW, LLW and SFIS (Ref. 46). I chose to assess the 8 operations that accrued the highest collective doses (see section 4.2.2.3) and which accounted for 55% of the planned exposure as identified by Hitachi-GE. These activities included:

- Reactor Opening / Closing Series Work.
- Reactor Well Decontamination.
- In-Service Inspection (ISI) Preparations Work in Drywell.
- Residual Heat Removal (RHR) Pump Inspection and Maintenance.
- Fine Motion Control Rod Drive (FMCRD) Replacement / Overhaul.
- Reactor Internal Pump (RIP) Motor Overhaul.
- Clean Up Water (CUW) Heat Exchanger Inspection and Maintenance.
- CUW Pump Inspection and Maintenance.

117. There are a significant number of other operations specifically planned or in response to breakdowns that accrued collective dose, however, as part of the Step 4 assessment (Ref. 3) only eight most significant from an exposure perspective were sampled.

4.2.2.1 The Hitachi-GE Approach to ALARP

118. It should be noted that the overall ALARP evaluation is covered specifically in PCSR Chapter 28 (Ref. 47), so this section will focus on the effectiveness of the Hitachi-GE approach in delivering an ALARP outcome for occupational exposure.

119. The Hitachi-GE approach to ALARP in GDA is described in (Ref. 47). The application of this approach to occupational exposure is described in (Ref: 46).

120. These documents were reviewed and it was found that the Hitachi-GE ALARP process draws heavily on ONR, HSE and other documentation such as (Ref. 9), (Ref. 49 and Ref. 50), with Hitachi-GE employing a process based on the use of good practice, improved where reasonable practicable to do so, as illustrated by the figure below, taken from (Ref. 47).

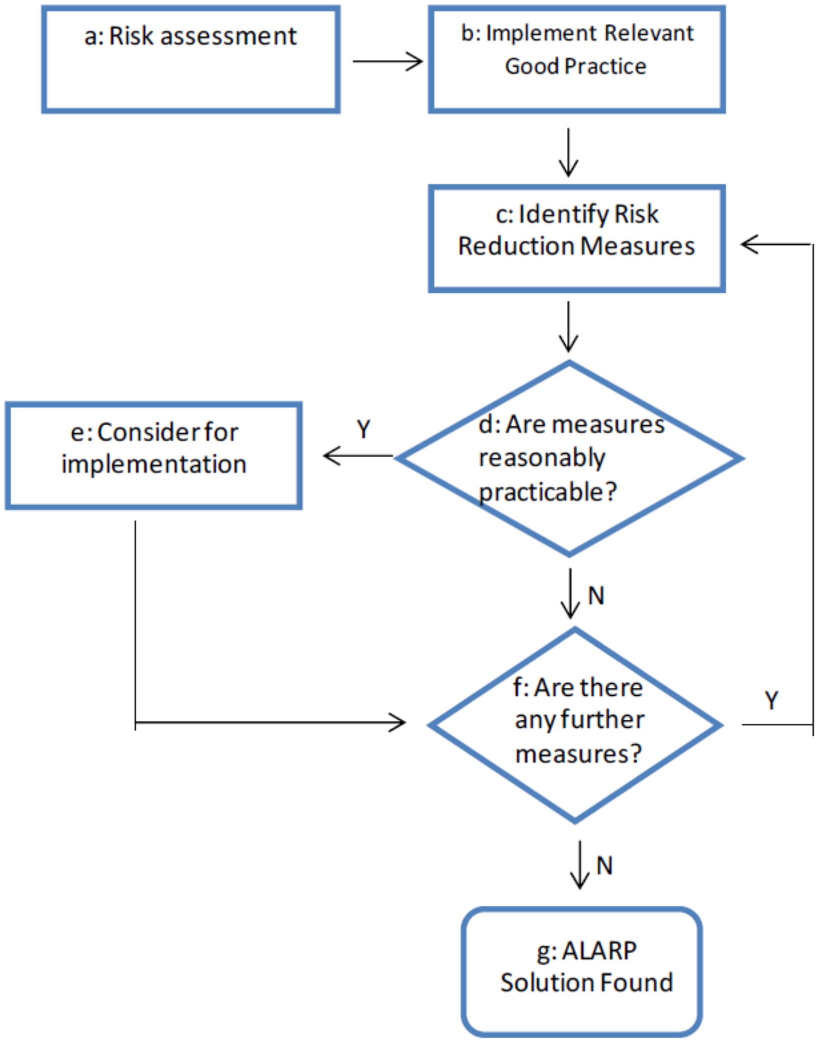

Figure 2: Hitachi-GE ALARP Process Flow Diagram (Ref. 47).

121. From a radiological protection perspective, the methodology presented is conceptually reasonable and consistent with UK standards. Shortfalls were however identified when the methodology was developed and applied in detail, notably in Contamination Control and Protection against Direct Radiation (Ref. 51). It was found that whilst collective dose over the plant lifetime versus difficulty of implementation may be a useful aid in making ALARP judgements, the dose bands used must be relevant to the task being considered. Collective dose bands that may reasonably be applied at plant level, as per Topic Report: Demonstration to Ensure that External and Internal Doses are ALARP (Ref. 46) (i.e. up to 2 man-Sv, 2 man-Sv to 20 man-Sv and over 20 man-Sv) are not meaningful when applied at system or task level as they will effectively screen out options that may in fact be reasonably practicable and reflect Relevant Good Practice (RGP).

122. Review of this document led to RQ-ABWR-0960 - Application of ALARP Methodology at Task Level (Ref. 52) on the application of the “traffic light” approach to qualitative ALARP assessment detailed in section 7.1 of GDA ALARP Methodology (Ref. 48). This resulted in Hitachi-GE changing its methodology for occupational exposure, as described in the RQ response UK ABWR ALARP Demonstration Methodology (Ref. 53). I considered the new approach to be appropriate.

4.2.2.2 ALARP at a System Design Level

123. I reviewed a number of system reports on a sampling basis to determine the effectiveness of the Hitachi-GE approach to ALARP. It should be noted that many details on the implementation of ALARP in general are contained in Contamination Control and Protection against Direct Radiation (Ref. 54), which was provided in response to RO-ABWR-0064. See section 4.4.2 for more details.

124. The BDL was subject to considerable review by ONR assessors from radiological protection, reactor chemistry and structural integrity disciplines to determine whether the design reduced risks, including occupational exposure, to ALARP, which included raising RO-ABWR-0034. Detailed analysis and assessment of this system demonstrated that the design was ALARP. Further detail of the review and conclusions is given in section 4.5.1.

125. A number of other areas of system design were also considered as follows:

- Topic Report on ALARP Assessment for Solid Waste Management System (Ref. 55).
- Topic Report on ALARP Assessment for Liquid Waste Management System (Ref. 56).
- Topic Report on the ALARP Assessment for the R/A, T/B, Rw/B, S/B HVAC Systems (Ref. 57).
- Topic Report on ALARP for Off Gas Systems (Ref. 58).

126. I reviewed these reports and concluded that although the HVAC, radioactive waste and Off-Gas systems have been through ALARP review and modifications have been identified and implemented, further work on ALARP demonstration is required. For example:

- Regarding the document on HVAC systems, I concluded that a statement on decommissioning activity would need developing before the end of Step 4 as a baseline assessment should be made prior to completion of Step 4. The current document is clear that the HVAC has not currently been designed for decommissioning.
- Noting that this is an ALARP assessment of the systems identified in paragraph 125, I concluded that there is no reference to minimising the activities requiring worker intervention, e.g. inspection work, to minimise the number of persons at risk and time at risk and therefore minimising the hazard prior to engineering controls.
- In the Topic Report on ALARP for the Off-Gas System, I concluded that there is evidence that options have been identified that may reduce dose, but as yet, there is no demonstration of how these have been applied to generate an ALARP solution.

127. In conclusion, I have raised an Assessment Finding:

AF-ABWR-RP-01: *The licensee shall ensure the appropriate application of ALARP with respect to the GDA design of Solid and Liquid waste management, HVAC and Off-Gas systems. This shall include optimising these systems for decommissioning activity, minimising worker interventions for maintenance where reasonably practicable to do so and fully evaluating options identified in Topic Reports, such that the site specific design is optimised and risks, including radiological risks, to workers are reduced so far as is reasonably practicable.*

4.2.2.3 Process ALARP

128. This section covers significant aspects of the process ALARP case for the highest dose operations as detailed in Contamination Control and Protection against Direct Radiation (Ref. 54).

129. Reactor opening and closing operations were described in GDA ALARP Demonstration for All Risks During RPV Top Head and FMCRD Removal (Ref. 59), with further information on minimisation of occupational exposure given in (Ref. 54). As this latter document was raised in response to RO-ABWR-0064, a fuller description of the aspects it addressed is given in Section 4.4.2 of this document. Major ALARP aspects, particularly related to external occupational exposure are discussed here.

130. A number of RQs were raised by ONR associated with these operations. These were principally associated with the technology employed to remove the head, working practices within the reactor well, steps taken to minimise contamination of the reactor well and control of water level within the RPV.

131. There are a number of potential technologies available to remove the RPV head of a BWR. This was discussed with Hitachi-GE in detail with RQ-ABWR-1065 (Ref. 60) raised to elicit a formal response on their view of RGP for this task, being followed up with RQ-ABWR-1312 (Ref. 61). Hitachi-GE provided a comprehensive response, however it was agreed that this required further work during the licensing phase. As a result, I have raised an Assessment Finding (AF-ABWR-RP-02).

132. One of the areas discussed with Hitachi-GE with regard to reactor well decontamination was optimum surface finishes for the reactor well. Surface finishes is in fact a much broader issue which was led by ONR assessors in the Reactor Chemistry discipline and is discussed further in Section 4.4.2.

133. Control of water level within the RPV has an impact on occupational exposure due to the shielding given by the water to contaminated reactor furniture, particularly the dryer, which is the uppermost element. In response to RQ-ABWR-1245 (Ref. 62), which included questions on this matter, Hitachi-GE identified a number of options which could improve control of the water level. These have also been included within the following Assessment Finding:

AF-ABWR-RP-02: *To ensure adequate demonstration of ALARP the licensee shall determine the equipment and operational process required for RPV head removal and reseating that reduces radiological protection risks SFAIRP. This requires assessing options for automation, e.g. use of Multi Stud Tensioning (MST) devices. Along with this the licensee shall examine reasonably practicable options that enable the RPV to be filled with water and maintained at a higher level than currently achieved in J-ABWR, prior to removal of the RPV head to ensure dose rates are ALARP.*

134. Another significant radiological protection issue is ALARP options for the removal of the dryer and separator from the RPV. It was concluded that the originally proposed arrangement was sub-optimal and it has been revised as described in the update to Ref. 63. It was reported that “The reasonably practicable option was to provide a Control and Instrumentation (C&I) protection system which ensures the Separator remains suitably submerged during handling. The risk reduction achieved by handling the Dryer submerged has been assessed to outweigh considerably any disadvantages of submerging the RBC hook (i.e. potential contamination of the hook or SFP water).”

135. I raised RQ-ABWR-1311 (Ref. 64) to ask further questions about the FMCRD handling machine and FMCRD maintenance activity. More detail on this is given in section 4.4.2 on RO-ABWR-0064.

4.2.2.4 Conclusion

136. I concluded that the Hitachi-GE approach to ALARP was conceptually sound and was consistent with UK legislation and ONR principles and guidance. However, it became clear that there were shortfalls in the application of the process. These have been addressed by Hitachi-GE, nevertheless there are a number of aspects of UK ABWR design and operation that need further substantiation from the perspective of occupational exposure. These have been captured as Assessment Findings (AF-ABWR-RP-01 and AF-ABWR-RP-02).

4.2.3 Normal Operation – Designation of Areas

137. PCSR chapter 20.5 (Ref. 31) provides information regarding radiation and contamination zoning for the UK ABWR. The main body of text specifies the boundary and limits for radiological and contamination zoning. Additional information for the assessment is appropriately referenced within this section.

138. Part of the step 4 plan (Ref. 3) was to review and assess the radiation and contamination zoning for the UK ABWR; I assessed the following aspects against UK legal requirements and relevant good practice:

- Radiation and Contamination Zoning.
- Area classification.
- Zoning Maps.
- Worker Activities.

139. As stated in section 2, Nuclear Technologies TÜV SÜD was contracted to review and assess aspects of designation of areas; their report (Ref. 65) shall be used in my assessment.

4.2.3.1 UK Legal Requirements

140. Designation of areas within a new nuclear facility is a key design aspect and is required within the UK under the IRR99 (Ref. 14).

141. IRR99 (Ref. 14) reg. 16 (Designation of controlled or supervised areas) stipulates that an employee shall designate areas as controlled or supervised areas based on the effective dose a worker shall receive over the calendar year, or if special procedures are required to restrict significant exposure.

142. Other aspects of IRR99 (Ref. 14) are also applicable; reg. 8 (Restriction of exposure) where an employer shall take all reasonably practicable steps to restrict exposure to employees on a site (hierarchy of control). It should be noted that within reg. 7 (Prior risk assessment) a risk assessment must be completed before any work involving ionising radiation may be undertaken.

143. The SAP’s (Ref. 6) provide further clarity to help guide my assessment for the proposed new nuclear facilities. RP.3 stipulates that where appropriate, designated areas should be further divided, with associate controls to restrict exposure and prevent the spread of radioactive material. RP.4 stipulates that effective means for protecting persons entering and working in contaminated areas should be provided.

144. Further requirements are stated within the radiation protection TAG (Ref. 10) para. 5.6. The zone category should be an indication of the required degree of engineered and managerial controls and should increase e.g. C1, C2, C3 and R1, R2, R3, etc., dependent on the increase in radiological and contamination risk. In para 5.7 it stipulates that access to low radiation zones should not require workers to go through a higher radiation zone. Instead that higher category zones be nested within less highly categorised zones.

4.2.3.2 Radiation and Contamination Zoning

145. Hitachi-GE has provided appropriate documentation (topic reports, supporting documentation and RQ responses) detailing the methodology behind the radiological and contamination zoning for the UK ABWR.

146. Hitachi-GE has completed a review of existing UK good practice regarding radiological and contamination zoning (Ref. 66). This has incorporated reviewing other licensees' approach to zoning, international guidance and experience from Suitably Qualified and Experienced Persons (SQEP) Radiological Protection Advisers (RPAs).

147. From this review Hitachi-GE has proposed the following radiological classification zoning for the UK ABWR (Table 3).

IRR 99	Radiation Classification	Dose Rate (μSv/h)
Un-Designated Area	R0	Less than 2.5
Supervised Area	R1	2.5 to 7.5
Controlled Area	R2	7.5 to 50
	R3	50 to 500
	R4	More than 500

Table 3: The proposed Radiological Zoning classification for the UK ABWR.

148. From Table 3 the dose rates applied to the upper and lower boundaries for a supervised area (2.5 to 7.5 μSv/h) are similar to those stated in IRR85 (Ref. 67), which was the precursor to the IRR99 (Ref. 14). Although this range is based on IRR85 the general range applied is not inconsistent with IRR99 (Ref. 14).

149. However from reviewing area designations applied by UK Licensees operating NPP's it is noted the supervised area dose rates applied to the boundaries are lower. Therefore the supervised/controlled area boundary is lower. It should be noted the Guidance on the Demonstration of ALARP TAG (Ref. 9) states that for a new reactor design "the level of safety must be no less than a comparable facility already working or being constructed in the UK or somewhere else in the world".

150. In summary I assess the radiological zoning criterion for the UK ABWR to be generally appropriate and to meet the requirements of IRR99 (Ref. 14). I conclude that opportunity for future work is possible so as to better align the radiological criterion with other UK licensees and we note that the licensee will be required to review this during the site specific phase. A minor shortfall (MS-UKABWR-RP-01) has been raised.

151. **MS-UKABWR-RP-01:** *Opportunity exists to better align the designation of areas for the UK ABWR with Relevant Good practice during site licensing. The Licensee should review the current radiological zoning criteria (specifically for supervised areas) to be consistent with current UK RGP for operational NPP's.*

152. From this review the Hitachi-GE has proposed the following contamination classification zoning for the UK ABWR (Table 4).

IRR99	Contamination Classification	Criteria	Notes
Un-Designated	C0	An area not designated under the IRR99	Meet requirement of EPR 16 Schedule 23, Part 6 (Ref. 22)
Supervised Area	C1	An area where conditions shall be kept under regular review.	Less than C2 Limits
IRR99	**Contamination Classification**	**Radionuclide**	**Radioactivity Lower Limit**
Controlled Area	C2 (Surface Contamination)	Alpha emitters	0.4 Bq/cm^2
		Radionuclides not otherwise specified (including Tritium)	4 Bq/cm^2
		^{55}Fe, ^{59}Ni, ^{63}Ni, ^{99}Tc, ^{125}Sb or ^{144}Ce (low energy beta emitters)	40 Bq/cm^2
	C3 (Airborne Contamination)	Where Alpha emitting radionuclides may be disregarded	Beta Limit – 10 Bq/m^3
		Where Alpha and Beta emitting radionuclides are present.	Alpha Limit – 0.01Bq/m^3
			Beta Limit – 2 Bq/m^3
		Where Tritium is present, and Alpha and Beta emitting radionuclides may be disregarded.	Tritium Beta Limit – 1 x 10^4 Bq/m^3
	C4	100 greater than C3 lower level limits	

Table 4: The proposed Contamination Zoning classification for the UK ABWR.

155. From Table 4 the contamination classification is similar to current good practice within the UK licensee.

156. For areas which are not designated under IRR99 (Ref. 14) the Hitachi-GE use the criteria set out in EPR 16 (Ref. 22).

157. For supervised areas, Hitachi-GE uses a similar approach to other UK Licensees and as stated under IRR99 (Ref. 14) these areas will be kept under regular review. However it is noted that Hitachi-GE has placed further criteria stating contamination shall be less than C2 criteria limits, this would imply the acceptance of contamination within C1 areas. My expectation is that the philosophy would be based around managing contamination levels in line with C0, recognising the potential for an increased risk of contamination at certain interfaces within the C1 area. However this does not preclude the need for an ALARP approach within the supervised areas. This has been raised as a minor shortfall.

MS-UKABWR-RP-02: *The licensee should review the approach applied to supervised areas with respect to contamination control so as to ensure any contamination is managed SFAIRP.*

158. For surface contamination classification (C2) in the UK-ABWR, Hitachi-GE has taken a similar approach to other UK Licensees. As seen in Table 4, the main difference is the low energy beta emitters which are based on the UK ABWR source term.

159. For airborne contamination (C3) there are some minor differences between the UK ABWR and other UK Licensees towards the lower limit levels for airborne contamination where the UK ABWR is more conservative with the limits. For the UK

ABWR the lower limit for alpha emitters when alpha and beta are present is set at 0.01 Bq/m^3, other UK Licensees have the same lower limit set at 0.015 Bq/m^3.

160. The reason for the difference is how Hitachi-GE has calculated the minimum levels for C3. Under ICRP 26 (Ref. 68) the Annual Limit of Intake (ALI) was defined as the quantity of radionuclide if inhaled or ingested in a year would result in that individual receiving an annual dose limit; under IRR99 this is 20 mSv (Ref. 14). It should be noted that under recent publication (ICRP 103 (Ref. 33)) ALI is no longer defined; however the methodology is a helpful tool. The ALI can be used for calculating the limits for inhalation via the derived airborne concentration; this is the concentration of activity of radionuclides in air which, when inhaled, over the course of a working year, would lead to the ALI. Hitachi-GE has utilised this relationship so as to determine the minimum C3 levels: 1/20th of the ALI value, e.g. 1 mSv.

161. Hitachi-GE use the dose co-efficient published in ICRP 68 "Dose Coefficients for intakes of Radionuclides by Workers" (Ref. 69). From reviewing the methodology used the approach undertaken to calculate the minimum safe levels (1 mSv as defined by Hitachi-GE) appears to be appropriate.

162. It is also noted that there is a discrepancy between the tritium levels for C3 controlled areas. For the UK ABWR the beta limit is $1X10^4$ Bq/m^3 whilst other UK Licensees are set at $6X10^4$ Bq/m^3.

163. For C4 levels again a similar approach has been undertaken to other UK Licensees. It should be noted that not all licensees within the UK have a classification system C0 to C4. Although the UK ABWR has a C4 classification, as shall be explained in section 4.2.3.4 currently no part of the UK ABWR zonal maps have worker access to C4 areas.

164. From reviewing the information provided the contamination zoning classification criteria for the UK ABWR is in line with current RGP within the UK Nuclear Industry and is fit for purpose.

4.2.3.3 Area Classification Methodology

165. Hitachi-GE took a five stage process to map the radiological and contamination zoning discussed in section 4.2.3.2 to the UK ABWR design.

- Use of KK-6/7 radiation and contamination zoning as a starting point.
- Application of design modifications from KK-6/7.
- Radiation and Contamination zoning taking into account design modifications from KK-6/7.
- Demonstration to ensure that external and internal doses are ALARP for each operation and maintenance activity.
- Final radiation and contamination zoning is determined taking into account ALARP assessment results.

166. It should be noted that the point relating to 'Demonstration to ensure that external and internal doses are ALARP for each operation and maintenance activity' is reviewed in more detail in section 4.2.3.5.

167. The reactors KK-6 and KK-7 are used by Hitachi-GE as a starting point to map radiological and contamination zoning for the UK ABWR. These two reactors have been chosen as they have reasonable quantities of relevant OPEX information available (in relation to worker dose/occupancy in areas), compared to other ABWRs KK-6/7 have been operating for a significant amount of time. Another reason for choosing these reactors is due to some similarity in certain aspects of water chemistry compared to the UK ABWR (specifically the iron control in KK-7).

168. Where design modifications have been undertaken for the UK ABWR in relation to the reference design e.g. KK-6/7, conversion factors are applied which take into account water chemistry and material selection. This allows dose rates to be estimated for each room using the EUST and applies to start-up, power operation and shutdown modes. For outages the radiological zoning is based on KK-6/7 OPEX data which is adjusted for the UK ABWR through the application of appropriate conversion factors. However; if significant design changes have been undertaken and those design changes invalidate KK-6/7 OPEX data, Hitachi-GE apply the most restrictive EUST. This is considered appropriate for this stage of GDA. Occupancy of areas is also considered, though it is acknowledged that this is likely to be modified by future licensees through the identification of a detailed operation and maintenance program.

169. A similar process is applied in relation to the definition of zoning in relation to contamination. Where modifications to the design have been undertaken for UK ABWR in relation to the reference design (i.e. KK-6/7) conversion factors are applied which take into account water chemistry and material selection. From a contamination perspective covering all operational modes for the UK ABWR, values are estimated based on KK-6/7 OPEX, with appropriate adjustment being made, through application of conversion factors. Though if significant design changes have been undertaken such that KK-6/7 OPEX data would be invalid then Hitachi-GE apply the most restrictive EUST. This is considered appropriate for this stage of GDA.

170. It should be noted that the TSC raised an observation (Ref. 65) regarding the approach undertaken by Hitachi-GE for zoning areas of the UK ABWR for both radiation and contamination. Within the response to an RQ (RQ-ABWR-1020 (Ref. 70)), Hitachi-GE states the KK-6/7 OPEX data represents the maximum values that have been measured at KK-6/7, this may not necessarily represent the maximum potential values in each area. In addition where the EUST has been used, due to significant design change, then it may be fitting to use OPEX information supported by modelling. This has been raised as a minor shortfall.

MS-RP-UKABWR-03: *The Licensee should consider reviewing zoning areas which are based solely on relevant OPEX information and incorporate modelling to ascertain if appropriate zoning has been achieved.*

171. Following my review of Hitachi-GEs methodology used for identifying the zoning criteria for both radiation and contamination for the UK ABWR I consider the methodology applied to be appropriate for GDA. The methodology provides a clear process and provides for the use of OPEX and source term information where required.

172. In addition to this methodology I also note that an ALARP assessment of both radiation and contamination zoning was undertaken by Hitachi-GE to assess that public and worker doses are ALARP (Ref. 71). The assessment was undertaken using the ERIC-PD model where zoning is considered as a dose reduction measure.

4.2.3.4 Zoning Maps

173. Hitachi-GE has provided zoning maps within its topic report (Ref. 71). A zoning map has been provided for start-up / power operation / shutdown and another map for outages (maintenance). The Zoning maps covered the following areas:

- Reactor Building.
- Turbine Building.
- Control Building.
- RadWaste Building.
- Condensate Storage Tank (CST).
- Suppression Pool Water Surge Tank (SPT).

174. From reviewing the available information in the topic report (Ref. 71) and from the TSC report (Ref. 65) the proposed reactor building zoning during operational phase for both radiation and contamination zoning is reasonable. The zoning maps align with the OPEX and EUST information provided.

175. For the reactor building during an outage there are several rooms where it appears that the classification from a contamination perspective is lower than OPEX data suggests. However Hitachi-GE responded to this observation (Ref. 72) by stating that during outage the contamination levels will increase for maintenance work, this is for a short term and when averaged over the day it is within the limits stipulated. This is common practice within the UK Nuclear industry and hence the information presented is appropriate for the reactor building.

176. For the turbine building during the operational and outage phases the radiation and contamination zoning appears to be reasonable. However it should be noted that one of the main rooms within the turbine building area has multiple radiation classifications but no physical barriers to prevent operators from entering these areas. When this observation was raised with Hitachi-GE, the response (Ref. 73) stipulated that it is common practice in other BWR's to have administrative controls (i.e. operator training / demarcated line). However within the UK it is RGP to have an engineered control as primary choice before the use of administrative control be undertaken. Although suggestions have been made for appropriate engineered controls it will be up to a future licensee to review this aspect to be in line with RGP within the UK; this has been raised as a minor shortfall (MS-UKABWR-RP-04).

MS-UKABWR-RP-04: *The Licensee should review the zoning arrangements in the Turbine Building and identify whether additional engineering controls are required to restrict access between zones.*

177. It should be noted that for both the Reactor and Turbine buildings the corridors and stairs are classed as C1 during operational terms though during an outage this is increased to C2. This could imply a reduction of standards in contamination control. Hitachi-GE stated that the maps should be considered to represent bounding radiation and contamination classification in order to reflect the maximum potential dose rate and contamination levels. It was also noted that it will be up to the future licensee to review and optimise the zone maps.

178. The majority of the control building is R0C0 (this is due to the control room) with the only area of significant radiation and contamination hazard being the main steam pipe line which is R4C2 during operation. From assessing information provided appropriate shielding is in place between the control room and main steam pipe; therefore the control building zone maps are appropriate for GDA.

179. For the Radwaste building there are several aspects which are still in a concept design phase (these relate to ILW and LLW) and shall be determined by the future licensee. Consequently the future licensee will provide appropriate radiological and contamination zoning for these areas.

180. Hitachi-GE states that the radiation and contamination zoning for the CST and SPT are appropriate for all operational phases of the UK ABWR.

181. Throughout the zoning maps provided the maximum radiation classification was set as R4 whilst for contamination the highest classification used was C3.

182. It should be noted that no zoning maps have been supplied for the Service Building; however Hitachi-GE provided information (Ref.54) on rooms that shall be inside the building. As with the Radwaste building the future licensee will provide appropriate radiological and contamination zoning for the service building.

183. As several of the buildings are in a concept design phase at the end of GDA (mainly the service building and parts of the Radwaste building). It shall be up to the future licensee to appropriately provide radiological and contamination zoning to these areas which follow the appropriate guidelines. As stated previously within this section several other aspects regarding zone maps will also be optimised at a later stage; this has been raised as a minor shortfall.

MS-UKABWR-RP-05: *As Hitachi-GE has not provided all the GDA radiological and contamination zoning for the nuclear island (specifically the service building and parts of the Radwaste building), the licensee shall provide appropriate radiological and contamination zoning for all relevant buildings for the UK ABWR.*

184. The TSC noted (Ref. 65) that it is common practice within UK Licensed Sites to produce colour contoured zoning maps depending on radiation and contamination classification (i.e. R1 areas shaded in green, R2 areas in orange). This approach is undertaken to help operators prevent ‘islands’ forming (low classification areas that are surrounded by high classification areas); this approach helps reinforce para 5.7 in the radiation protection TAG (Ref. 10). From reviewing Hitachi-GE’s zoning maps (Ref. 71) a colour coded approach has not been undertaken. Although from the assessment no ‘islands’ are visible it shall be up to a future licensee to consider implementing a colour coded zoning system; this has been raised as a minor shortfall (MS-UKABWR-RP-06).

MS-UKABWR-RP-06: *As it is common practice to have a colour- coded zoning maps, the Licensee should consider implementing a colour coded zoning maps to help prevent radiation and contamination classification ‘islands’ from forming.*

4.2.3.5 Specific Worker Activities

185. As part of the closure for RO-ABWR-0064 (see section 4.4.2), Hitachi-GE had to provide specific worker activities and the appropriate radiation and contamination zoning for both operational and outage phases. This was to clarify to ONR that Hitachi-GE had an appropriate understanding of zoning. These worker activities were:

- Transfer Cask loading and dispatch with radioactive materials.
- HEPA filter change.
- Maintenance and / or inspection of equipment.
- Change-room use for works in combined areas.
- Sampling material transfer.
- Radioactive waste handling and transfer.

186. For radioactive materials loading transfer cask/cask loading, and HEPA filter change, the information provided detailing the contamination zoning levels for each of the worker activities was based on OPEX information. From the review of the information provided in my opinion this appears to be in line my expectations.

187. For maintenance and / or inspection of equipment one of the areas discussed is the laundry drain system. This has been given a contamination classification of C3, this was in my opinion an inappropriate level as it was expected to be a maximum of C2. The Hitachi-GE responded to this observation (Ref. 73) and stated that the laundry drain system is currently in a concept design phase which will need to be appropriately reviewed by a future licensee. Application of a C3 classification is deemed as a conservative measure to give a future licensee the confidence that contamination control measures would be available if required.

188. Other maintenance / or inspection of equipment facilities sampled (Radwaste building, CST and SPT) are in alignment with GDA expectations.

189. Where change-rooms are provided in the form of sub-change-rooms to allow for access to controlled areas these have been given a C2 classification; I assess this as fit for purpose. It should be noted that Hitachi-GE provides appropriate information in Ref. 54 on sub-change areas for the UK ABWR.

190. For sampling material transfer in my opinion an appropriate approach is undertaken where samples are taken and transferred through C2 areas and not through a higher contamination zone area.

191. For radioactive waste handling and transfer in my opinion an appropriate approach is undertaken (the area is designated C2 and is supported by engineering and design layout in line with the hierarchy of control measures).

4.2.3.6 Conclusion

192. The radiological and contamination zoning criteria chosen for the UK ABWR is generally appropriate and follows similar RGP of other UK Licensees. Hitachi-GE has reviewed and appropriately created radiation and contamination boundaries which follow the regulatory requirement.

193. The five stage approach that Hitachi-GE has undertaken to map the radiological and contamination zoning criteria to the UK ABWR design is adequate. Hitachi-GE has made use of appropriate OPEX information from Japanese ABWRs, along with the EUST to calculate the zoning criteria for each of the UK ABWR areas. A minor shortfall has been raised regarding the use of combined EUST and OPEX data where applicable.

194. After reviewing the zoning diagrams I have concluded the majority of buildings for the UK ABWR generic layout have appropriate zoning classification provided for the varying status of the plant.

195. Overall I found the radiation and contamination zoning for the UK ABWR to be adequate; several MS have been raised.

4.2.4 Radiation Shielding

4.2.4.1 Introduction

196. Although in the UK there is no specific legislation governing the requirements and acceptability of shielding, the utilisation of effective shielding is a key control measure for restricting the exposure of personnel and the public. As a passive engineering measure, it follows the minimisation of radiation source terms in the hierarchy of control measures (Ref. 15). As a result, I have considered shielding design to be a principal aspect of my assessment.

197. The shielding assessment was undertaken to assess the UK ABWR shielding provisions identified in the PCSR submission (Ref. 31), to review the arguments presented in the PCSR, and to assess whether the evidence presented substantiated those arguments for shielding. The objectives of the shielding assessment were as follows.

- To be satisfied that the UK ABWR shielding design fulfilled the requirements outlined in the SAPs (Ref. 6), in particular RP.6 and in the TAG for radiation shielding (Ref. 11).
- To be satisfied that relevant good practice had been applied to the shielding provisions to support the demonstration that external dose rates and dose accrual by workers and members of the public were ALARP, taking into account international guidance from the IAEA (Ref. 35).

198. I was assisted in my assessment by ONR's specialist TSC, TÜV SÜD Nuclear Technologies (Ref.44). The information and references provided by Hitachi-GE in support of the PCSR as well as additional information provided in response to RQs were used as the basis for the assessment.

4.2.4.2 Documents Submitted For Review

199. In addition to the PCSR (Ref. 31) several other key documents were provided to support the safety case for shielding provision. These included:

- Topic Report: Public Dose Evaluation from Direct Radiation for All Relevant Buildings, ILW, LLW and SFIS during System Start-up, Power Operation, Normal Hot Stand-by, System Shutdown and Outages (Ref. 74).
- Topic Report: Radiation Shielding for all Relevant Buildings during System Start-up, Power Operation, Normal Hot Stand-by, System Shutdown and Outages excluding ILW, LLW and SFIS (Ref. 75).
- Public and Worker Dose Evaluation, Zoning and Radiation Shielding of SFIS System during Normal Operation (Ref. 76).

200. To support these documents, Hitachi-GE also issued the following documentation on shielding:

- Support Document: Physical Property of Radiation Shielding Materials (Ref. 77).
- Support Document: Dose Conversion Coefficients used in Radiation Shielding Calculation (Ref. 78).
- Support Document: Computer Codes used in Radiation Shielding Calculation (Ref. 79).
- Support Document: Radiation Shielding Calculation around Reactor Core (Ref. 80).
- Topic Report: Radiation Shielding for all Relevant Buildings during System Start-up, Power Operation, Normal Hot Stand-by, System Shutdown and Outages excluding ILW, LLW and SFIS (Ref.75).
- Support Document: Radiation Shielding Calculation around Radioactive Components (Ref. 81).
- Support Document: Radiation Shielding Calculation around Spent Fuel Pool Ref. 82).

201. Extensive documentation was also issued to describe and justify source terms and this informed the shielding review. This is described in Section 4.2.1 of this report and so is not considered further here.

202. Numerous responses were also issued in response to RQs raised.

4.2.4.3 TSC Review Scope

203. The technical review of the above documentation (Ref. 65) carried out by the TSC focussed on the following areas:

- **Source Terms:** High-level review of radionuclide inventory derivation. Selective review of gamma emission spectrum derivation from radionuclide inventories.
- **Shielding Design Basis Data:** Review of physical data (e.g. material densities and compositions, flux to dose conversion factors) used as the basis for calculations.
- **Calculation Methods:** Review of logic and methodology, key assumptions, computational codes and their adequacy for use in analysis.

- **Application of ALARP to Shielding Provision:** Review of design integration of operational experience and optimisation exercises in order to demonstrate that shielding provision is appropriate and that calculated doses to workers are ALARP.

4.2.4.4 TSC Review of Source Terms

204. A detailed technical review of the full suite of source term documentation was outside the scope of the work carried out by the TSC and further information on source terms can be found in section 4.2.1. Nevertheless, the TSC performed a high-level review of the documentation (Ref. 65) in order to ensure that the most significant sources of radiation with respect to reactor shielding design were taken into account and were derived in a logical and methodical manner, including:

- That the process of interpreting OPEX data (data from operational facilities) and the application of that data within the context of the UK ABWR is appropriate.
- That where OPEX data might not be considered appropriate, modelling or alternative means have been used to generate appropriate data.
- That all major contributors to the source term are appropriately accounted for, e.g. crud-burst on shutdown and start-up, fuel failures, presence of tramp uranium, variations in coolant chemistry.
- That the lists of radionuclides contain all of the expected radionuclides with respect to external radiation.
- The transformation from radionuclide fingerprints to gamma emission spectra for use in shielding calculations.

205. The TSC noted (Ref. 65) that the DST, unlike the PrST was not derived from the PST but was based mostly on operational experience and calculation. It concluded that overall, the volume and variety of reference data and how it had been modified and compiled for the UK ABWR helped ensure that the DST is adequately conservative from a radiation shielding perspective.

206. The TSC raised a number of queries via ONR on the subject of source terms and these are described in section 4.2.1.1; adequate responses to these queries were received.

207. Key source terms were then selected by the TSC in order to perform a gamma spectrum derivation to compare with those used by Hitachi-GE in radiation shielding and direct dose calculations. These included:

- Condensate Storage Tank Process Source Term (power operation, design basis).
- Condensate Storage Tank Deposit Source Term (power operation, design basis).
- Turbine Building (T/B) High Pressure Turbine Process Source Term, power operation, best estimate.
- Reactor Clean-Up Water System (CUW), PrST, design basis.
- CUW, DST, design basis.
- Condensate Purification System (CPS), PrST, design basis.
- Residual Heat Removal System (RHR), DST, design basis.
- Spent Sludge System (SS), DST, design basis.

208. In most instances, the TSC found good correlation between the spectra calculated by themselves and the spectra produced by Hitachi-GE, however minor differences were noted in the CUW and SS spectra and there was a difference in approach to the treatment of bremsstrahlung radiation. Following discussion with Hitachi-GE, the TSC concluded that Hitachi-GE approach was reasonable.

4.2.4.5 TSC Review of Shielding Design Basis Data

209. The review performed by the TSC (Ref. 65) considered the physical properties of the radiation shielding materials used for UK ABWR radiation shielding calculations. The assumptions outlined in the support document concerning physical properties of radiation shielding materials (Ref. 77) were compared to typical shielding material compositions and densities used within the UK nuclear industry.

210. The TSC (Ref. 65) found that in the majority of instances, material compositions were typical of those used for shielding calculations within the UK.

211. A query was raised regarding the composition of a material used in one application, known as “RSW mortar”, specifically its sensitivity to drying out over time, leading to reduced hydrogen content and hence lower effectiveness as a neutron shield. Hitachi-GE demonstrated in this instance that the dose-rate was dominated by gamma radiation and so the effect was not significant. The TSC was satisfied with this response.

212. The TSC examined fluence to dose conversion factors used in shielding codes and found that they were taken from the most relevant ICRP publication (ICRP 116) (Ref. 83).

4.2.4.6 TSC Review of Calculation Methods

213. The TSC found that the Monte Carlo radiation transport code MCNP5 version 1.6 was used for all shielding assessments for UK ABWR.

214. MCNP is produced by Los Alamos National Laboratory. Its wide use within the nuclear industry over many years has subjected it to numerous validation studies and comparisons with benchmarking experiments and it is internationally recognised for its capability to simulate neutron and gamma radiation transport in complex geometries.

215. The TSC (Ref. 65) considered that the selection of MCNP5 Version 1.60 for the radiation shielding calculations relating to the UK ABWR was acceptable as it is capable of simulating the shielding problems expected within UK ABWR. Additionally, MCNP is used widely within the UK (and worldwide) and is accepted as RGP.

216. The TSC (Ref. 65) then examined the application of MCNP5 by Hitachi-GE considering the documentation submitted and the confidence checks the TSC had undertaken. The TSC then concluded the following:

- Geometry modelling assumptions are suitably conservative.
- The cross-section data used for shielding calculations are adequate.
- Suitable treatment has been afforded to dose point selection on a case by case basis.
- Monte Carlo results are quoted with an acceptable relative error.

217. The TSC also concluded that although cross-checks were not carried out for every calculation, the quality assurance process followed by Hitachi-GE provides a degree of confidence that should spurious results arise, adequate investigation and independent calculations will be undertaken.

218. The TSC also examined the use of scaling and inference by Hitachi-GE in order to simplify calculations and found that within the available examples, the logic and assumptions are generally appropriate. Following response to queries raised they concluded that provided other assessments where scaling and inference have been used follow similar logic to those examined, they had no further concern with regards to the application of scaling and inference by Hitachi-GE.

219. In reviewing the calculations, the TSC noted (Ref. 65) that whilst much of the detail for radiation shielding calculations was presented within the reviewed documentation, reference to the document presenting the calculations was not made and the data was not brought together in one succinct location. This was raised by ONR in RQ-ABWR-1224 (Ref. 84), however this has not yet been satisfactorily addressed hence it will be taken forward as a minor shortfall.

MS-UKABWR-RP-07: *The Licensee should consider presenting detailed shielding data in one reference document, in order to facilitate shielding calculations during the site specific phase, to help prevent omissions within the licensee's documentation.*

4.2.4.7 TSC Review of Application of ALARP to Shielding Provision

220. The radiation shielding design approach outlined by Hitachi-GE is presented in Ref. 75, but is summarised as follows:

Step 1 - Kashiwazaki Kariwa units 6 & 7 (KK-6/7) were selected as the reference plant for the UK ABWR.
Step 2 - KK-6/7 shielding thicknesses were taken as the starting point.
Step 3 - Radiation shielding calculations were performed using the UK ABWR source term and KK-6/7 shielding thicknesses, with all relevant design modifications from KK-6/7 being taken into account.
Step 4 - The calculated dose rates were compared to the defined radiation zones.
Step 5 - If the calculated dose rates were within the respective design criteria, then it was concluded that the radiation thickness was sufficient to meet the radiation zone dose rate criteria.
Step 6 - If they were not, then the thickness of the shielding should be revised.

221. The TSC concluded (Ref. 65) that this process, followed from Step 1 through to Step 5 is analogous to common practice within the UK for determining preliminary bulk shielding thicknesses for a facility, where the design is sufficiently immature such that openings, doorways, labyrinths and penetrations may not yet be specified.

222. The original GDA Step 4 design submission for TSC review contained very little detail with regards to openings and penetrations. This was the subject of a number of queries raised by the TSC via ONR.

223. These queries ultimately prompted Hitachi-GE to issue a further document: The UK ABWR Penetrations Design Rule (Ref. 86).

224. The TSC considered that whilst Ref. 75 does not provide comprehensive assessment of all openings and penetrations, it does provide examples of shielding assessments of some key penetrations. ONR considers further work is required to adequately substantiate the claim that doses from all openings and penetrations are reduced SFAIRP. ONR raised RQ-ABWR-1222 (Ref. 85) to obtain evidence to support Hitachi-GE's claim, however ONR assessed the response as not providing adequate evidence. This is the subject of Assessment Finding AF-ABWR-RP-07. See section 4.4.3.7 for further detail.

225. In other respects however, the TSC considered that the practice implemented by Hitachi-GE concerning the specification of shielding is in line with common international practice and demonstrates the potential to yield personnel doses which can be considered to be ALARP.

4.2.4.8 TSC Review Conclusions

226. The conclusions of the TSC (Ref. 65) review were as follows:

- The clarity, presentation, self-consistency and referencing of the documentation supplied by Hitachi-GE is generally of a high standard.
- Evidence of an iterative design process with an associated practical document control system has been provided.
- Calculations have been performed using established methods and verified computational codes.
- Radionuclide source inventories have been converted to gamma energy spectra with adequate conservatism.
- Geometries, material compositions and assumptions are adequately conservative.
- Flux to dose conversion factors are appropriate.

227. Note: Where the TSC has replicated calculations reported by Hitachi-GE, results and conclusions have been in agreement.

4.2.4.9 RQs Raised Resulting from ONR Assessment and TSC Review

228. I assessed the UK ABWR documentation providing the shielding safety case on a sampling basis in addition to the review carried out by the TSC. This prompted me to raise a number of RQs, in addition to the RQs prompted by queries from the TSC. These RQs covered a range of topics as indicated below:

RQ ID	Main Technical Area	Related Technical Area	Summary
RQ-ABWR-0158 (Ref 87)	Radiation Protection & (Level 3 PSA)	Civil Engineering	Approach to shielding design
RQ-ABWR-0174 (Ref. 88)	Radiation Protection & (Level 3 PSA)	Radwaste & Decommissioning	Radiation protection aspects associated with Spent Fuel Interim Storage (SFIS)
RQ-ABWR-0902 (Ref.89)	Radiation Protection & (Level 3 PSA)	PSA	Supply of additional information relating to Hitachi-GE SQE Personnel Knowledge of Computer Shielding Modelling Codes
RQ-ABWR-1019 (Ref.90)	Radiation Protection & (Level 3 PSA)	Reactor Chemistry	Direct Dose to the Public
RQ-ABWR-1068 (Ref. 91)	Radiation Protection & (Level 3 PSA)	Generic Environmental Permitting	Public Dose From Direct Radiation Assumptions
RQ-ABWR-1124 (Ref. 92)	Radiation Protection & (Level 3 PSA)	Generic Environmental Permitting	Assumptions made for Calculating Direct Shine from the Turbine Building
RQ-ABWR-1133 (Ref. 93)	Radiation Protection & (Level 3 PSA)	None.	Source Data Regarding Prompt Fission Gamma Rays and Neutrons
RQ-ABWR-1222 (Ref. 85)	Radiation Protection & (Level 3 PSA)	None.	ALARP demonstration with regards to Radiation Shielding

RQ-ABWR-1224 (Ref. 84)	Radiation Protection & (Level 3 PSA)	Radwaste & Decommissioning	Radiation Shielding
RQ-ABWR-1226 (Ref. 94)	Radiation Protection & (Level 3 PSA)	Generic Environmental Permitting	Skyshine Contributions to Public Dose
RQ-ABWR-1232 (Ref. 95)	Radiation Protection & (Level 3 PSA)	Reactor Chemistry	Radiological Protection OPEX
RQ-ABWR-1245 (Ref. 62)	Radiation Protection & (Level 3 PSA)	Radwaste & Decommissioning	Contamination Control and Protection Against Direct Radiation - Control of Exposure During RPV Head Removal

Table 5: List of RQs relating to Radiological Shielding.

229. The majority of these RQs were addressed to my satisfaction, however as discussed above, further work is required to fully address RQ-ABWR-1222 and RQ-ABWR-1224 (Ref. 85 and 84).

4.2.4.10 Conclusion

230. Overall, the standard of documentation received in support of GDA has been adequate to allow a sufficiently detailed examination of the UK ABWR bulk shielding design and some aspects of detailed shielding provisions.

231. All of the documentation submitted by Hitachi-GE and made available to the shielding TSC at the time of review, along with the responses to the RQs issued to Hitachi-GE demonstrate that the shielding design is being developed through a logical and iterative design process using acceptable methods, shielding codes and adequately conservative assumptions.

232. The documentation submitted by Hitachi-GE supporting the UK ABWR shielding design demonstrates that, when reviewed in the context of the guidance and expectations outlined in the SAPs (Ref. 6) and TAG (Ref. 11) the shielding provisions are acceptable. I have identified no reason why the shielding design of the UK ABWR will not be capable of reducing external dose rates so far as is reasonably practicable.

4.2.5 Monitoring of Radiation Exposure

233. The previous section (Section 4.2.4) of this report, reviewed shielding provisions and protection against external radiation which could lead to external exposures. During this section I shall review the provision of monitoring for both external and internal radiation. Chapter 20, Sub-chapter 20.6 (Ref. 31) provides information relating to the monitoring provisions for radiation and contamination within the generic design of the UK ABWR, specifically for employees working with Ionising Radiations and other employees on site. The section also provides a brief summary of the monitoring provisions in relation to monitoring of public exposures.

234. My Step 4 assessment plan identified the need to assess Chapter 20.6 (Ref. 31) and supporting documentation including the Topic Report on Radiation and Contamination Monitoring of Occupational Exposure (Ref. 96) and relevant references.

235. These sections of the Safety Case are assessed against the following standards associated with the monitoring of radiation and contamination in relation to occupational exposure:

- UK Legal & Regulatory Requirements, primarily IRR99.
- ONR guidance.

- Relevant Good Practice.

236. My assessment specifically focuses on how Hitachi-GE identifies and defines its basis for monitoring within UK-ABWR and so considers:

- Methodology used to identify the requirements for monitoring of contamination and radiation including use of fault schedules to identify primary issues to guard against.
- Use of optioneering to identify the means of monitoring and relay of information to relevant Human Machine Interfaces.
- Identification of the relevant parameters to be monitored ensuring appropriate specification of equipment and the measurement/operational constraints required.

4.2.5.1 UK Legal Requirements

237. The main regulations which are applicable to Radiological Protection and against which I assess the proposed safety case are IRR99. The main regulations within this which relate to control of radiation exposures are:

- Regulation 8 (Ref. 14) Restriction of exposure: establishes a hierarchy of control measures. Monitoring and installed monitoring forms part of this hierarchy. This hierarchy includes warning devices which alert operators to faults or failures which have occurred and which reduce the safety integrity of the installation. These devices do not prevent exposure but will indicate to the operators what action to take and not to take. Portable monitoring provides support to administrative controls.
- Regulation 10 Maintenance and examination of engineering controls etc., and personal protective equipment.
- Regulation 11 Dose Limitation including Dose Limits for members of the public and other persons.
- Regulation 18 (6) Significant risk of spreading contamination, specifically the provision of monitoring for contamination any person, article or goods leaving a controlled area.
- Regulation 19 Monitoring of designated areas.

238. Further to this I have considered:

- The Management of Health & Safety at Work Regulations (MHSWR) (Ref. 97) Regulation 5
    - (1) Every employer shall make and give effect to such arrangements as are appropriate, having regard to the nature of his activities and the size of his undertaking, for the effective planning, organisation, control, monitoring and review of the preventive and protective measures.
    - (2) Where the employer employs five or more employees, he shall record the arrangements referred to in paragraph (1).

4.2.5.2 ONR Guidance

239. NS-TAST-GD-038, Rev 6, Radiological Protection (Ref 10), and NS-TAST-GD-043, Rev 3, Radiological Analysis – Normal Operation (Ref 12), provide further advice and guidance to inspectors in relation to application of IRR99 (Ref. 14) and Nuclear Site Licence requirements.

4.2.5.3 Methodology for Radiation and Contamination Monitoring of Occupational Exposure

240. During Step 4 Hitachi-GE presented information detailed earlier in this section. Early reviews of this documentation led to a number of questions being raised in level 4 meetings, the responses to which along with the originating documents, provided confidence in the arrangements for monitoring radiological conditions in the generic UK ABWR design. When considering this information, and the fact that the selection of instrumentation for radiological monitoring will be the responsibility of the future licensee, I decided not to focus significant assessment effort in this area. This is consistent with previous GDA assessments and is an example of ONRs proportionate approach.

241. Hitachi-GE states the Area Radiation Monitoring system (ARM) provides direct radiation monitoring information within accessible operational areas, in support of safety and is distinct from the Process Radiation Monitoring system (PRM), which is primarily focused to support plant process control.

242. The Monitoring framework consists of two methodologies which complement each other. These are:

- Trend Monitoring where the radiological condition of a designated area and the measurement trend is recorded over time. Appropriate action levels will be set by the licensee prior to commissioning. Trend Monitoring is intended to inform plant management of changes in operational circumstance and conditions and allow corrective actions to be taken. This may include:
  - Determining where detailed surveys may be required.
  - Increaseing frequency of accurate measurement.
  - Change on-going or planned systems of work.
- Accurate measurement through detection and reporting of more localised and directed measurements of radiation or contamination values. The data can be used to determine effective dose which will inform decisions on work planning. Generally this is provided by portable equipment read and handled by trained health physics personnel.

243. No options are precluded by the proposals for design generated by Hitachi-GE in relation to the UK ABWR. Certain parts of the generic design are in concept design phase and in general the principles applied give confidence that an adequate level of protection can be achieved.

244. It is noted the systems related to trend monitoring are to be supported by alternating current uninterruptible power supply (UPS). Providing a level of resilience in relation to potential loss of power.

245. Decommissioning was specifically excluded from Draft B of the PCSR. However, I have sampled areas of the decommissioning safety case (Ref. 98) and supporting reports to provide confidence that they do not preclude development of monitoring provisions.

246. Hitachi-GE considers adequacy of space and accessibility in relation to the need for calibration and maintenance ensuring safe access. I have not carried out a detailed assessment of the exposures related to maintenance and calibration of installed radiation monitoring equipment as the exact location and nature and extent of calibration and maintenance will be defined by the licensee. Hitachi-GE state that measurements from installed monitors will be communicated to the Control Room and back up Control Rooms.

247. External (outdoor) Installed Area monitoring equipment is designed to detect and record photon ambient dose equivalent rates at the site boundary. The focus is to monitor potential dose to the public, present at the site boundary, along with a primary purpose of warning to operators of change in operational circumstance. Positioning of the detectors is to be carried out in the site specific phase. As with internally installed equipment all data is relayed back to the MCR and displayed with appropriate alarm panels. The system is supported by the Emergency Diesel Generators in case there is a loss of off-site power.

248. Equipment is designed to meet the appropriate measurement ranges associated with design basis accident conditions and meet relevant outdoor environment conditions in relation to, temperature, humidity, etc. This is consistent with my expectations for GDA in line with ONR guidance and relevant good practice. It is expected that the Licensee will further define both locations and nature of the external area monitoring equipment.

249. Revision 3 of the Topic Report (Ref. 96) has seen its title extended to cover “Part of Basis of Safety Cases on Other Control and Instrumentation System” This has been driven by C&I requirements.

250. Further to this and following work on prevention of access to High Dose and Dose Rate Areas (Ref. 99) in line with the requirements of RO-ABWR-0064, Hitachi-GE recognises the need to consider provision of access control interlocks which are driven in line with process control as well as measured radiation and contamination levels for the protection of the workforce. This is now acknowledged within the document and safety case (Ref.31).

251. Hitachi-GE claims the use of portable monitors meets the requirements of the nuclear industry ventilation design guide ES_0_1738_1 Ventilation Systems for Radiological Facilities (Ref. 100). This claim is not fully supported as the document specifically states “The health physics aspects of activity monitoring are not part of this document, but its objectives place demands on ventilation systems design.” The guide goes on to state “Airborne activity levels will normally have to be measured in all occupied GREEN and AMBER areas of buildings. This may require the installation of sampling equipment for air contamination measurements. For facilities such as reactor plant, where significant levels of alpha emitting nuclides are unusual, air sampling with retrospective assessment will probably suffice.” This would imply samplers providing retrospective measurement capability may be used. Since the standard clearly denotes GREEN areas are equivalent of C2 as identified by Hitachi-GE and applied to UKABWR areas, the expectation would be samplers are included in the design as a minimum within C2 areas to monitor activity in air trends. This is particularly important during commissioning and through into full operation.

252. The reference plant on which the UK ABWR design is based has limited provision for installed area radiation monitoring system (ARM) linked to interlock systems or warning devices to inform or prevent inadvertent access to areas of potential high dose or contamination e.g. HVAC/off-gas filter rooms or liquid waste handling rooms. This is a shortfall and an assessment finding has been raised (AF-ABWR-RP-03).

AF-ABWR-RP-03: *The licensee is required to review the GDA requirement for the use of installed activity in air and direct radiation monitoring, such that appropriate locations are identified and the design facilitates adequate detection and signalling of data to relevant interlocks and warning devices to ensure exposure of workers is controlled so far as is reasonably practicable.*

253. The Ranges for installed Radiation monitoring equipment identified as a minimum for the UK ABWR Area Radiation Monitoring System are detailed in Table 18 of the Topic Report (Ref. 96) These are:

Radiation Zone			Minimum Acceptable Range µSv/h
Designation		Radiation Level µSv/h	
Undesignated	R0	<2.5	$10^{-1} – 10^{2}$
Supervised	R1	2.5 - 7.5	$10^{-1} – 10^{4}$
Controlled	R2	7.5 - $5x10^{1}$	$10^{-1} – 10^{4}$
	R3	$5x10^{1}$ - $5x10^{2}$	$1 – 10^{4}$
	R4	>$5x10^{2}$	$10 – 10^{4}$

Table 6: Minimum acceptance range for installed radiation monitoring equipment for each radiological zone.

254. There is no requirement for criticality monitoring due to the nature of the fuel storage and handling design. This is fully assessed in the Fuel & Core report (Ref. 16).

255. Although Hitachi-GE has not identified any installed activity in air monitors/samplers specific to monitor the occupational exposure, airborne activity is monitored for plant performance by the PRM . The PRM primary function is plant process control. For example the airborne activity is monitored within the Primary Containment Vessel (PCV), drywell. The PCV monitoring is provided to indicate a loss of containment from the Reactor Pressure Vessel (RPV) into the PCV where worker access during power operation is prevented.

4.2.5.4 Conclusion

256. From the information provided, I have no significant concerns with regard to the ARM system and the Generic Design for Radiological Instrumentation. The primary means of identifying incidents is related to the PRM system. Hitachi-GE has provided installed instrumentation to monitor for increased ambient dose rates which provide early warning to the operator. My assessment has not examined the doses involved with maintaining and calibrating the equipment because these factors will depend on the exact specification and location of the instruments, and this will be determined at the site specific phase. The ability to interrogate instrumentation remotely supports the optimisation of exposure by minimising the need to routinely access instrumentation.

257. There is one specific assessment finding associated with radiological instrumentation as identified in the assessment above. This will need to be addressed satisfactorily by the future licensee.

4.2.6 Normal Operation – Public Exposure

258. Section 1.5 of my Step 4 Plan (Ref. 3) explained that my assessment of public exposure would include the following matters.

- Liaison with EA on behalf of NRW on optimisation of doses to the public from direct radiation originating within the site boundary (ONR has the lead).
- Liaison with the EA on behalf of NRW along with Liabilities Management Regulation Inspectors (LMR) on optimisation of doses to the public from authorised discharges (the EA has the lead).

259. I was assisted in the review by ONR's specialist TSC, TÜV SÜD Nuclear Technologies and the agreed objectives of my review were to ensure that:

- The UK ABWR NPP design fulfils requirements outlined in UK regulatory documentation: IRR99 (Ref. 14), SAPs in particular RP.6 (Ref. 6) and the TAG for Radiation Shielding (Ref. 11).
- To be satisfied that relevant good practice had been applied to the shielding provisions to help to demonstrate that external dose to members of the public will be ALARP taking into account relevant national and international guidance.

4.2.6.1 Documents Submitted for Review

260. In addition to the PCSR (Ref. 31) a number of other key documents were provided in support of the safety case for direct dose to the public. These included:

- Topic Report: Public Dose Evaluation from Direct Radiation for All Relevant Buildings, ILW, LLW and SFIS during System Start-up, Power Operation, Normal Hot Stand-by, System Shutdown and Outages, Rev 3 (Ref. 74) and Rev 2 (Ref. 101).
- Topic Report: Demonstration to Ensure that External and Internal Doses are ALARP for All Relevant Buildings during System Start-up, Power Operation, Normal Hot Stand-by, System Shutdown and Outages excluding ILW, LLW, SFIS, Rev 2 (Ref. 46).
- Public and Worker Dose Evaluation, Zoning and Radiation Shielding of SFIS System during Normal Operation (Methodology and Example) (Ref. 76).

261. As detailed in section 4.2.4 on Radiation Shielding extensive documentation was also issued to describe and justify source terms. This informed the shielding review and also this review into public dose.

262. As explained in Section 2.1.3 above, the regulation of public radiation exposure during normal operation is shared between the EA and ONR, where IRR99 (Ref. 14) is enforced by ONR and EPR16 (Ref. 22) is enforced by the EA. IRR99 (Ref. 14) requires dose constraints to restrict exposure to ionising radiation at the planning stage where it is appropriate to do so. The guidance to IRR99 (Ref. 15) advises that a constraint for a single new source should not exceed 0.3 mSv per year for members of the public. This is repeated in the SAPs (Ref. 6) in relation to NT.1 Target 3 (see Table 7), and advises that ONR's view is that a single source should be interpreted as a site under a single duty holder's control, since this is an entity for which radiological protection can optimised as a whole. However, the PHE-CRCE recommended that the dose constraint for members of the public from new NPPs should be 0.15 mSv per year (Ref. 23).

Normal operation – any person off the site	**Target 3**
The target and a legal limit for effective dose in a calendar year for any person off the site from sources of ionising radiation originating on the site are: BSL(LL): 1 mSv BSO: 0.02 mSv *Note that there are other legal limits to tissues and parts of the body (IRR99).*	

Table 7: SAPs NT.1 Target 3.

263. Hitachi-GE adopted design criteria in the form of constraints, whereby dose from the sum of direct radiation from all buildings on the site to members of the public are less than the BSO of 0.02 mSv per year at the site boundary. The individual design criteria

are set out in Table 8 below. The total for the facility as a whole is 0.0195 mSv per year (19.5µSv per year).

Facility	Annual Dose Design Constraints at the Site Boundary (mSv per year)
Reactor Building (R/B)	0.0005
Turbine Building (T/B)	0.005
Radwaste Building (Rw/B)	0.0005
Condensate Storage Tank (CST)	0.005
Suppression Pool Water Surge Tank (SPT)	0.001
Spent Fuel Interim Storage (SFIS)	0.005
ILW/LLW Facilities Collectively	0.0025
Total	0.0195

Table 8: Hitachi-GE Design Criterion for Generic Design.

264. Hitachi-GE presents evidence in support of the performance of the UK ABWR and supporting facilities design in the Topic Report on “Public Dose Evaluation” (Ref. 74). The report provides arguments and evidence to support its assertions relating to public exposure being controlled so far as is reasonably practicable and therefore doses are as low as reasonably practicable. Hitachi-GE has been unable to fully assess all of the relevant buildings due to the level of maturity of the design of those buildings reaching concept design stage. To date Hitachi-GE has provided detailed evaluations of the Reactor Building, Turbine Building, Radwaste Building, Condensate Storage Tank and Suppression Pool Water Surge Tank. Hitachi-GE has also provided a non-refined estimate for the Spent Fuel Interim Storage facility. The results of this work are presented in Table 9.

Building	Distance: GDA Site Boundary (m)	Annual Dose*1 (microSv/y)
Main Buildings:	-	-
Reactor Building	265	1.30E-02
Turbine Building	310	8.82E-01
Radwaste Building	243	2.51E-02
Control Building	-	~0
Service Building	-	~0
Tanks:	-	-
Condensate Storage Tank	240	4.59E-03
Suppression Pool Water Surge Tank	350	1.74E-02
Spent Fuel Interim Storage:	120	(5.00E+00)*2
ILW / LLW:	-	-
ILW Storage Facility	90	(5.00E-01)*2
Wet ILW Processing Facility	280	(5.00E-01)*2
Solid ILW Processing Facility	210	(5.00E-01)*2
Wet LLW Processing Facility (Treatment Facility)	250	(5.00E-01)*2
Solid LLW Processing Facility and Temporary Storage Facility	160	(5.00E-01)*2
Total	-	8.44E+00

*1: This annual dose is based on the BE source term.

*2: These values are set to be dose constraints because these facilities are still at the concept design stage.

Table 9: Public Dose from each Building in at GDA Site Boundary (Ref. 74).

265. As can be seen from the results of Hitachi-GEs dose assessment to date presented in Table 9 the performance against Target 3 can be expected be below the BSO 20 μSvy^{-1} (Ref. 6). It is noted a number of facilities still require detailed assessment, although use of design constraints in the assessment provides assurance direct dose to the public can be maintained well below the BSO. This is accomplished as a result of the deployment of effective shielding around the reactor, its containment and more specifically the Turbine Building and Suppression Pool Water Surge Tank.

266. The designs for a number of buildings in support of the GDA are not mature enough to apply a detailed model for assessment of doses to workers or the public. Although it is not necessary to have carried out detailed assessment at this point, Hitachi-GE has identified and committed to suitable dose constraints for planning purposes. Compliance with these dose constraints will need to be addressed by the Licensee during detailed site specific design.

4.2.6.2 TSC Review of Public Dose due to Direct Radiation

267. The TSC TÜV SÜD Nuclear Technologies completed a review of public dose due to direct radiation (Ref.103), the scope of review covered:

- Design Criteria: This considers the design criteria used by Hitachi-GE to determine the acceptability of the design with respect to the public dose evaluation from direct radiation.
- Shielding Design Basis Data: Review of source terms and physical data (e.g. material densities and compositions, flux to dose conversion factors) used as the basis for calculations.
- Calculations Methods: Review of logic and methodology, key assumptions, computational codes and their adequacy for use in assessments.

- Application of ALARP: Review of design integration of operational experience and optimisation exercises in order to demonstrate that calculated annual doses to members of the public from direct radiation are ALARP.

268. The TSC reviewed Hitachi-GEs submission in relation to the application of the UK regulatory requirements and the application of the hierarchy of protection/control.

269. After assessing the aspects which apply to protecting members of the public, those hazard reduction measures which are most appropriate include:

- Reduce the Hazard: for example by selecting appropriate materials for piping to minimise the activated corrosion/erosion product component of the coolant source term.
- Isolate the Hazard: for example by specifying adequate shielding provisions.
- Control the Hazard: for example by appropriate placement of the site boundary with respect to the most significant contributors to dose (e.g. Turbine Building)

270. TSCs review (Ref.103) was undertaken largely taking into account radiation shielding good practice. This is the primary means of isolating and controlling the hazard, following hazard reduction, which is the focus of other Topic Areas.

271. In a similar approach to that adopted in Section 4.2.4 on Radiation Shielding the TSC reviewed source terms at a high level and found that the derived gamma spectra used for calculations concerning the public dose evaluation from direct radiation are suitably conservative and are considered adequate for use. The TSC identified a number of observations which were raised during the assessment with ONR and communicated with Hitachi-GE via RQs. The responses to these RQs were adequate.

272. Following the responses a review of the shielding assessment was undertaken to inform the assessment of public dose to direct radiation. The most significant shielding material with respect to members of the public is provided by concrete as concrete forms the majority of the bulk shielding for all of the key buildings within the generic site envelope. No significant issues were identified through this review.

273. Further to this, the TSC carried out reviews of the flux to dose conversion factors used as inputs to the models along with the calculation and computational codes used. The TSC and ONR considered the use of independent verification within Hitachi-GE. This was achieved by the use of cross checking results by alternative computer codes or by analytical methods which confirmed that procedures and processes exist within Hitachi-GE that meet this expectation.

274. The TSC carried out independent calculations concerning key components that contribute to annual doses to members of the public from direct radiation; the CST and the T/B using the discrete ordinate code Attila. The calculations used the geometry, source terms and dose points provided by Hitachi-GE, but used assumptions and material compositions consistent with those used within the UK nuclear industry.

275. These comparison calculations identified that with respect to the CST, the DST was the more onerous source term and is shielded by the less active process water when the CST is full, dose rates at the site boundary being an order of magnitude less than one micro Sievert. Initial findings showed a partially full CST may yield higher dose rates than a full CST and the sensitivity analysis of this matter was performed and showed that a CST which is two thirds full yields a dose rate at the site boundary of up to approximately a factor of 3.5 higher than a full CST. This observation was raised in an RQ with Hitachi-GE along with a similar query with respect to the SPT. Responses to these RQs and publication of revisions to the Topic Report on Public Dose Evaluation (Ref 74) and Source Term suite address these issues.

276. Further calculations were carried out with respect to the Turbine Building and Reactor Building. Detailed calculations were carried out for the Turbine Building. This led to a number of RQs being raised to seek clarification with respect to differences in calculated results being observed. Following responses to the RQs the TSC confirmed the adequate closure of its observations. A less complex assessment of the Reactor Building was modelled by Hitachi-GE with the core being modelled as an homogenous sphere. Initial investigations and comparisons by the TSC using this method generated a number of areas of discrepancy and clarification was sought through RQs. The response from Hitachi-GE again provided additional clarification on a number of points which duly satisfied the TSC. The responses included a revised model using a cylindrical form for the reactor building. The resultant skyshine component was suitably lower than the spherical model by two orders of magnitude, so demonstrating the conservative nature of the spherical model. The TSC considered the responses to the RQs suitable to close the observations raised.

277. The TSC investigated the general treatment of skyshine and due to the responses to RQs considered Hitachi-GE's approach to be suitably conservative.

278. The TSC review of Spent Fuel Interim Storage and ILW/LLW facilities identified that detailed dose assessments have not been carried out for these facilities at this time. A preliminary example and methodology report (Ref. 76) has been issued regarding the SFIS facility, but a similar report regarding the ILW and LLW facilities remains as work in progress at this time. The example and methodology report for the SFIS concluded that further work would be required to demonstrate that the dose rates from direct radiation at the site boundary would be acceptable. The need to address the shielding issues with respect to the SFIS is an important finding; however as the facility is in a concept design phase this will be reviewed during licensing.

279. The TSC considered the application of ALARP through this process. The initial step in consideration of ALARP was to consider the application of worst case, unmitigated annual dose at the site boundary as is common practice within the UK. This is carried out to provide early indication of a problem with the site meeting the design criteria or legal requirements. This unmitigated dose can then be reduced by considering factors such as average operating conditions rather than chronic worst-case, more realistic occupancy assumptions for members of the public at the site boundary, actual locations of residence and shielding factors for time spent indoors versus time spent outdoors.

280. Although the methodology employed by Hitachi-GE considered all such variables in order to demonstrate the annual dose to a member of the public at the site boundary would be significantly below the design criterion of 19.5 µSv per year, it did not apply the worst case.

281. The TSC carried out a worst case calculation using the unmitigated dose at the site boundary. This revealed the total annual design basis unmitigated dose at the worst case location at the site boundary from the R/B, T/B, Rw/B, CST, and SPT was 3.6 µSv per year. This is less than the combined design criterion for these buildings of 12 µSv per year. If the completed concept designs for the SFIS and ILW/LLW facilities meet their total design criteria of 7.5 µSv per year, the total annual design basis unmitigated dose of 11.1 µSv per year would still be well below the total design criteria of 19.5 µSv per year.

282. An example of further qualitative justifications which can be outlined regarding why there is confidence that the maximum annual dose accrued by members of the public from direct radiation will be significantly less than calculated, is the assumption of the reactor being operational for a full year without an outage.

283. The TSC carried out further assessment of the application of ALARP through this process considering the relevant use of OPEX and learning from experience. The TSC and ONR raised RQs seeking clarification with regard to certain aspects of shielding development and identified that the roof of the T/B is thicker than the reference plant KK-6/7.

284. Following the update of the Topic Report on Public Dose Evaluation (Ref 75) the TSC and ONR note that further work could be done with respect to ALARP evaluations of the CST and SPT areas. This is seen as a minor shortfall considering the position in respect to GDA and the work still to be carried out during site specific phase.

MS-UKABWR-RP-08: *The Licensee should further develop the ALARP evaluation for the CST and SPT to better demonstrate that there is adequate shielding to reduce public dose SFAIRP.*

285. It should be noted that direct radiation is only one component of the dose to members of the public from the operation of the UK ABWR. Exposures will also come from both liquid and gaseous discharges and these are estimated to be in the order of 23.8 µSv per year to the most exposed individual (Ref. 104). To this end efforts to reduce liquid and gaseous discharge may be more effective in reducing doses to the public than measures to reduce direct dose.

286. Informed by the assessment work carried out by the TSC I consider, notwithstanding the work still required to be competed with regard to the SFIS, LLW/ILW, that there is no reason to suspect that the design criteria for the above facilities will not be achieved by their respective concept design (5 µSv and 2.5 µSv per year respectively) (Ref. 74). My opinion is based on my confidence in the methods applied and responses received thus far by Hitachi-GE.

287. Throughout Step 4, radiological protection assessors and radioactive waste and decommissioning assessors from ONR have jointly attended meetings with assessors from the EA on topics which have common interest, such as radioactive waste, decommissioning and decontamination. As such, I have liaised with the EA on matters regarding public doses which are outlined in its assessment report (Ref. 105).

4.2.7 Normal Operation – Radiation Dose for Work Activities

288. PCSR sub-chapter 20.8 (Ref. 31) provides information regarding worker dose assessment for the UK ABWR, specifically for employees working with ionising radiation and other employees on the site. Information on the worker dose methodology along with an example of a high dose activity is provided. Additional information for the assessment is appropriately referenced within this section.

289. Part of the Step 4 plan (Ref. 3) was to review and assess the following aspects of worker dose for the UK ABWR:

- Exposure to workers of highest individual annual dose for assessment against target 1.
- Highest annual group average dose in respect to target 2.

290. These were assessed via looking at the following areas in relation to worker dose and assessing them against UK legal requirements and relevant good practice:

- Methodology for Worker Dose (both for employers working with ionising radiation and other employers working on the site).
- Specific Worker Dose Activities.

291. Although this section reviews the worker dose and methodology for calculating the collective and maximum individual dose, more information regarding the ALARP assessment for worker activities is provided in section 4.2.2.

4.2.7.1 UK Legal Requirements

292. Dose Limitation within a new nuclear facility is required within the GB under the IRR99 (Ref. 14).

293. IRR99 (Ref. 14) reg. 11 (Dose Limitation) stipulates that every employer shall ensure that his employees and other persons who are on site do not exceed dose limits specified in Schedule 4.

	Dose Limits (mSvy^{-1})		
	Employee Over 18	**Employee Under 18**	**Other Persons**
Effective dose	20	6	1
Lens of the eye	150	50	15
Skin (averaged over 1cm^2)	500	150	50

Table 10: IRR99 Schedule 4 dose limits (Ref. 14).

294. The SAPs (Ref. 6) provide further information regarding dose targets for new nuclear facilities against regulatory requirements and relevant good practice. NT.1 provides numerical targets and limits which a safety case should be assessed against. These targets include the Basic Safety Level (BSL) which is sometimes the legal limit, and the Basic Safety Objective (BSO). The BSO is at a point where further assessment via ONR is deemed an unreasonable use of resource; however if ALARP measures are still valid to a licensee below the BSO, these must be undertaken by law.

295. Tables 11 and 12 provide the BSL and BSO limits for both employers on site and other workers on site as well as any group on site.

Normal operation – any person on the site	**Target 1**
The targets and a legal limit for effective dose in a calendar year for any person on the site from sources of ionising radiation are: Employees working with ionising radiation: BSL(LL): 20 mSv BSO: 1 mSv Other employees on the site: BSL: 2 mSv BSO: 0.1 mSv *Note that there are other legal limits on doses for specific groups of people, tissues and parts of the body (IRR99). Normal operational doses should also be reduced ALARP.*	

Table 11: SAPs NT.1 Target 1.

Normal operation – any group on the site	Target 2
The targets for average effective dose in a calendar year to defined groups of employees working with ionising radiation are: BSL: 10 mSv BSO: 0.5 mSv	

Table 12: SAPs NT.1 Target 2.

296. The Radiological Analysis for Normal Operation TAG (Ref. 12) provides further information regarding the targets stipulated within Tables 11 and 12. Within Para 5.21 it states that a group of workers should not be 'diluted' with workers who receive very low doses that significantly reduce the average dose to the group.

297. For high dose activities in para 5.22 it stipulates that they should have been analysed and the need for engineered provisions included in the design as there may be tasks that could give rise to relatively high doses to specific workers.

298. Both of the above targets discussed in Tables 11 and 12 are relevant and consistent with IAEA safety standards (Ref. 106); specifically to the following fundamental principles outlined in the IAEA safety standard:

- Optimisation of protection - Protection must be optimised to provide the highest level of safety that can reasonably be achieved.
- Limitation of risks to individuals - Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.
- Protection of present and future generations - People and the environment, present and future, must be protected against radiation risks.

299. For the assessment of worker dose these aspects along with the targets stipulated within the SAPs shall form the bases of my assessment on worker dose.

4.2.7.2 Methodology for Worker Dose

300. Hitachi-GE took a seven stage approach when preparing the UK ABWR dose evaluation for employers working with ionising radiation (Ref. 107).

- Selection of Reference Dose.
- Prioritisation of Worker Activities based on Reference Dose.
- Identification of 'Mitigating Options'.
- Implementation of MO's and Selection of Reasonably Practicable Options.
- Estimation of UK ABWR dose.
- UK ABWR collective dose target / individual Dose Constraint.
- Overall ALARP Review.

301. For the selection of reference dose, Hitachi-GE reviewed occupational exposure from national and international reactors around the world. The average ABWR occupational exposure was demonstrated as the best performing BWR worldwide.

302. The UK ABWR is referenced on a specific ABWR; currently several Japanese ABWR's are operational. One of the main criteria for the reference plant is regarding the main source term: ^{60}Co, which has a half-life of 5 years. For the first 5 years of UK ABWR operation, dose rates from deposition of ^{60}Co continue to increase. After the initial 5 years it is expected that dose rates and occupational exposure should reach equilibrium. Currently only two ABWR's have operated for over 5 years; Kashiwazaki-Kariwa Units 6 and 7 (KK-6/7). As the iron control for the UK ABWR is similar in design to KK-7, this was chosen as the reference plant for dose. Additional worker dose

information was provided regarding KK-6. The process for choosing the reference plant as discussed is appropriate for GDA scope.

303. As KK-7 is the reference plant collective dose data was used to help calculate the UK ABWR. Information from its 5th, 6th, 7th and 9th periodical inspection was used; the 8th was not part of the assessment due to the effects of the Niigataken Chuetsu-oki Earthquake and therefore this period was not considered representative.

304. From the review of maintenance a sample of target activities was chosen for more detailed analysis to help with worker dose assessment. These consisted of the eight high dose activities (equivalent to about 50% of total collective dose) and seven low dose activities. The sample chosen is appropriate and more information on these choices is discussed later.

305. A risk assessment was undertaken detailing for each of the above high dose activities the relevant design features and administrative controls to minimise radiation and contamination. These have been based on mitigating options that use the ERIC-PD tool to deliver the application of the hierarchy of control. These mitigating options were then reviewed to ascertain if a 'reasonably practicable' option can be applied to reduce radiation and contamination risk further. Further information on this aspect is discussed in section 4.2.2.

306. For estimating the UK ABWR dose from the above information a three stage process was undertaken:

- KK-7 reference dose and relevant OPEX information.
- Estimating the conversion factors from the reference dose.
- Apply the conversions factors.

307. The conversion factors to be applied relate to different water chemistry and material selection that the design of the UK ABWR shall implement compared to KK-7. Other conversion factors to be applied relate to changes in the UK ABWR design compared to the reference design; these are:

- Upper component of the FMCRD will be maintenance free.
- Bottom Drain Line (water chemistry and material selection of BDL).
- Shortening of CUW piping within upper drywell.
- Application of Reliability Centred Maintenance (RCM).

The conversion factors were applied to KK-7 values to identify the UK ABWR average collective dose.

308. From reviewing the information provided and the process for estimating the UK ABWR dose, questions were raised regarding the conversion factor approach. Information provided (Ref. 46) stipulated how conversion factors affected the average collective dose for the UK ABWR compared to the reference plant KK-7 for both operational and periodical inspection plant status. It was noted that operational collective dose had been unaffected by conversion factor changes. As stated already one of the conversion factors is the application of RCM which minimises maintenance frequency through appropriate monitoring of equipment and associated operational parameters. RCM would have an effect on collective dose during operation status of the UK ABWR. However no dose reduction estimates are provided for RCM as Hitachi-GE stipulates this will be considered in more detail at a licensee stage, when the operational programme will have been set. A similar case applies to a further conversion factor associated with shortening of the CUW piping where detailed pipework design is to be completed at site license stage.

309. Due to the incomplete development of conversion factors providing appropriate dose reduction information the estimated collective dose for the UK ABWR provided is only a partial representation and will require further work. I have raised a minor shortfall (MS-UKABWR-RP-09) in relation to this point.

310. It is RGP for a licensee to set a collective dose/individual dose target/constraint for a reactor. Hitachi-GE reviewed relevant OPEX information from international BWRs as well as Japanese BWRs / ABWRs to arrive at the UK ABWR target (Ref. 107). The targets set (see Table 13) are comparable to the best operating reactors and hence fit for purpose.

311. The final stage of the dose methodology is to compare the collective dose target / individual dose constraint to the estimated UK ABWR dose to see if further dose reduction techniques are achievable and comparable to each other.

UK ABWR	Target	Estimation
Collective Dose per Year (Person-mSv)	500	501
Maximum Individual Dose per Year (mSv)	10	11

Table 13: Comparison with UK ABWR Dose target and Estimated Dose (Ref. 107).

312. From Table 13 the average collective dose per year for the UK ABWR is nearly in line with the UK ABWR target/constraint. As stated previously the UK ABWR collective dose is only a partial representation. From a GDA perspective the UK ABWR collective dose per year is fit for purpose.

313. From Table 13 the maximum individual dose per year is above the current target. Although this is within the legal limit stipulated in Table 11, further work still has to be completed by a future licensee to ensure the UK ABWR constraint is met; this will be addressed through application of MS-UKABWR-RP-09.

314. When assessing the worker dose assessment for appropriate conservatism there appears to be some omissions. Firstly when calculating the estimated UK ABWR Collective Dose Hitachi-GE has used the average of the four collective doses (this is the 5th, 6th, 7th and 9th Periodical inspection from KK-7 OPEX data) and omitted repair work OPEX information (Ref. 107). It is reasonably foreseeable that during operation / outage there will be unplanned worker activities which will need to be resolved that will incur a collective dose. Although it is unplanned and hence a value cannot be provided it is still foreseeable to have a range on the estimated collective average dose.

315. Another aspect is the average collective dose range. When reviewing the collective dose for the same worker activity during the 5th, 6th, 7th and 9th periodical inspection for KK-7, it can be seen that in some cases there are significant variations in collective dose between each periodical inspection. When this was raised with Hitachi-GE, the response (Ref. 108) stipulated that this can be due to the frequency of the inspection or maintenance activity leading that different worker activities are conducted in different periodical inspection. An example is during one periodical inspection the CUW Regenerative heat exchanger is undergoing maintenance whilst the Non-Regenerative heat exchangers is haveing an inspection, whilst another periodical inspection it is vice versa. This can affect the average significantly and this was not taken into consideration for calculating the collective dose.

316. Considering the conversion factors used, as discussed previously RCM and shortening of the CUW pipeline have not been taken into account (this is due to the scope of

GDA); however this could be construed as potentially affecting the appropriate conservatism.

317. When considering if appropriate conservatism has been applied for calculating dose for employees working with ionising radiation, there appears to be areas where Hitachi-GE has opportunity to refine and improve upon the assessment.

318. The dose methodology for other employers on site is a similar process to employees working with ionising radiation; the following process occurs:

- Selection of Reference Design.
- Initial Evaluation.
- UK ABWR Individual Dose Constraint.
- Identification of Mitigation Options.
- Implementation of Mitigating Options and Selection of Reasonably Practicable Options.
- Overall ALARP review.

319. Hitachi-GE use KK-7 as the reference plant for other employees on site; public dose data is used to select KK-7 as the main difference between public dose and other employees on the site is down to occupancy as well as distance from radioactive sources. Public dose data provided show that direct radiation from KK-7 buildings are within natural background radiation level (Ref. 107) Further to this KK-7 has a greater length of operational service than other ABWR's for the same reason as stated earlier; I find this reason and selection of reference plant to be in line with GDA scope.

320. For initial evaluation Hitachi-GE undertook a computer simulation using appropriate Source Term information (EUST for radiation protection (Ref. 32)), and the location of source buildings as well as civil structure requirements. They modelled a dose from a distance of 30m from each building.

321. From the results a value of 48 μSvy^{-1} was calculated as the dose to other employees on the site (Ref. 107). It should be noted that this value is based on the best estimate source term. SFIS and ILW/LLW facilities are in concept design and have been assigned a dose constraint value.

322. However within the PCSR (Ref. 31) Hitachi-GE states a different value of 10 μSvy^{-1}. The reason for the difference is in relation to the inclusion of SFIS and ILW/LLW facilities within the worker dose topic report (Ref. 107).

323. The UK ABWR dose constraint for other employees on the site has been set as 100 μSvy^{-1}; this aligns with the BSO for any other employees on site.

324. As with employees working with ionising radiation, appropriate mitigation options have been chosen using the ERIC-PD tool.

325. The final stage was an overall ALARP review where Hitachi-GE challenged the design to see if any additional controls or modifications to the design can be reasonably applied to reduce the dose further. It should be noted that although the value provided is below the BSO for other employees on site, the future licensee will be required to carry out further ALARP reviews during the detailed design for the SFIS and ILW/LLW facilities; this is captured within MS-UKABWR-RP-09.

MS-UKABWR-RP-09: *The licensee should provide ONR with an appropriate estimate of the maximum individual and collective dose for the UK ABWR, taking into account updated conversion factors and other dose reduction tools, including appropriate consideration of conservatism as necessary in line with RGP. This should also include the facilities that were in a concept design at GDA stage (SFIS, LLW and ILW facilities).*

326. For other employees on site it should be noted the SFIS and LLW/ILW facilities are in concept design and have been provided with a dose constraint. From the sample of information reviewed it has not been possible to ascertain where these values have come from. Again this is linked MS-UKABWR-RP-09.

327. A worked example using the methodology described above was provided (Ref. 76) regarding worker dose (both for employees working with ionising radiation and other employees on site) of the SFIS system during Normal Operation.

328. Assumptions applied by Hitachi-GE in calculating dose to workers (Ref. 76) did not include a component for internal dose. Hitachi-GE state that during the spent fuel handling, transfer, export and storage there will be no potential for radioactive materials to leak during normal operation from the multi-purpose canister due to the welding of the lid. Although this may be the case, ONR would expect a licensee to demonstrate/provide further evidence of the little or no exposure for an internal dose to occur. This has led to a minor shortfall.

MS-UKABWR-RP-10: *The Licensee when completing dose assessment for all aspects of the UK ABWR should ensure appropriate account is taken for all components of exposure. Where a component is not believed to be significant, justification should be provided.*

329. For the example provided for worker dose assessment for working with ionising radiation the assessment states the dose is below the BSL but above the BSO. The document further goes on to state there are options available to further reduce worker dose (i.e. remote operations) though due to the SFIS being in concept design this shall be completed by the future licensee. A dose assessment is undertaken for dose to other employers on the site where the estimated dose is above the current constraint stipulated for the SFIS. This is related to a previous minor shortfall (MS-UKABWR-RP-09)

4.2.7.3 Worker Dose for Specific Tasks

330. As stated in section 4.2.2 as part of the dose methodology, Hitachi-GE provided information on the eight high dose worker activities within the UK ABWR reference design.

- Reactor Opening / Closing Series Work.
- Reactor Well Decontamination.
- ISI Preparation / Work In Drywell.
- RHR Pump Inspection and Maintenance.
- FMCRD Replacement / Overhaul.
- RIP Motor Overhaul.
- CUW Heat Exchanger Inspection and Maintenance.
- CUW Pump Inspection and Maintenance.

331. Hitachi-GE also provides examples of major worker activities within the turbine building.

- Main Turbine Inspection and Maintenance.
- Moisture Separator Reheater Inspection and Maintenance.

332. Hitachi-GE provided specific worker activities with the appropriate radiological and contamination zoning for both operational and outage phase. These worker activities were:

- Radioactive Materials loading transfer cask / cask loading and dispatch.
- HEPA filter change.
- Maintenance and / or inspection of equipment.
- Change-room use for works in combined areas.
- Sampling material transfer.
- Radioactive waste handling and transfer.

333. These have previously been discussed in section 4.2.3.5. With regards to the eight high dose worker activities, these shall be discussed in this section and provide an overview of the collective dose (ALARP demonstration for worker activities is discussed in section 4.2.2).

334. It should be noted that the average collective dose for KK-7 is based on the 5th, 6th, 7th and 9th periodical inspection; the 8th was not part of the assessment due to the effects of the Niigataken Chuetsu-oki Earthquake and therefore this period was not considered representative.

335. In the majority of the eight high dose worker activities sampled the UK ABWR dose has a similar value to the KK-7 average collective dose for worker activities. Dose reduction measures have been identified (Ref. 109), though as stated previously, the dose reduction measures ‘application of RCM’ and ‘shortening of the CUW pipeline’ require more detail design information which is out of scope from GDA.

336. The most significant reduction regarding the eight high dose worker activities is within the FMCRD maintenance; this is due to no planned Upper Component (U/C) maintenance during the lifespan of the UK ABWR. However the ability to remove the U/C will still be available.

337. Within these eight worker activities a more detailed review was undertaken for the RPV head installation / removal and reactor well decontamination. A review of the contamination approach for these two activities is discussed in section 4.4.2.6. Hitachi-GE has stated that more work on these worker activities shall be completed by the future licensee; this was noted and discussed within section 4.2.2.3 which led to an assessment finding (AF-ABWR-RP-02).

338. In summary Hitachi-GE identified a number of dose reduction measures which in principle support the optimisation of worker exposures and therefore reduce doses. The evidence supplied in support of the effectiveness of these dose reduction measures is limited and it is Hitachi-GEs expectation that this will be further supported during the site specific phase. As I would expect this to be developed in line with normal business for future site licensee’s to review worker dose for all worker activities I consider this a minor shortfall (MS-UKABWR-RP-09).

339. Hitachi-GE also provided information (Ref. 66) on the average and maximum individual dose to workers for each of the high dose activities over the 5th, 6th, 7th and 9th periodical inspection (Outages). From reviewing the information none were above the BSL (this is the legal limit of 20mSv) (NT.1 Target 1), but the majority were above the BSO of 1mSv. In a small number of cases the maximum individual dose exceeded the dose constraint set by RP of 10mSv. Consequently, a future licensee will be required to assess the maximum individual and collective doses, taking into account final design decisions, and demonstrate that they are ALARP, as required by MS-UKABWR-RP-09.

340. With regards to Target 2 for NT.1 (Normal operation – any group on site) from reviewing the above eight worker activities and comparing the average worker dose to the target half of the activities were above the BSO.

4.2.7.4 Conclusion

341. From reviewing information provided by Hitachi-GE relating to worker dose assessment, I am satisfied that the worker dose methodology for employees working with ionising radiation is appropriate. It takes a step by step approach, taking into account appropriate OPEX information as well as using conversion factors to take into account the different water chemistry the UK ABWR will have compared to the reference plant. Also the constraint set by Hitachi-GE for collective dose and maximum individual dose is in line with national and international guidance (Ref. 54). However it should be noted that some of the conversion factors are still to be reviewed at the site specific stage. With regards to the annual collective dose for the UK ABWR, Hitachi-GE's estimated value is comparable to the constraint, and as discussed, further dose reduction measures (i.e CUW shortening of the pipe / RCM) are to be reviewed at the site specific stage. The estimated employees individual dose is below the IRR99 legal limit of 20 mSv (Ref. 14) although it is above the BSO for target 1 as well as above the constraint Hitachi-GE has stipulated for a maximum individual dose to a worker.

342. From reviewing the information provided by Hitachi-GE relating to worker dose assessment I am satisfied that the methodology for calculating doses to other employees on the site is appropriate. A similar approach is undertaken to calculate dose as with employees working with ionising radiation, with the main difference being the use of computer modelling. The dose to other employees on site is below the BSO where Hitachi-GE's constraint is 0.1 mSv. My only concern relates to the concept design of the SFIS and the LLW and ILW stores with regards to the assumptions used to calculate the values. This again raises the issue on appropriate conservatism when calculating the estimated worker dose.

343. Regarding the worker dose assessment for the eight highest worker dose activities for the UK ABWR, in general dose reduction techniques have been identified, but due to information not being available at the GDA stage, the full potential of the dose reduction techniques are not available. Hence the doses for these worker activities have similar or marginal improvements to the performance of the reference plant. It is also noted from an earlier section section (4.2.2.3) that further work is required on some of the worker activities previously raised as an assessment finding (AF-ABWR-RP-02); this will again change the worker dose assessment. It is also apparent in the majority of worker activities that collective dose for a group is below the BSO set by Target 2 of NT.1.

344. From the above assessment I have raised two minor shortfalls. The first (MS-UKABWR-RP-09) addresses the points I raised in paragraphs 340 to 343, whilst the second minor shortfall (MS-UKABWR-RP-10) was raised to address the potential for omission regarding the calculations for worker dose.

4.2.8 Post-Accident Accessibility

345. Chapter 20.9 of the PCSR (Ref. 31) contains Hitachi-GEs claims, supporting arguments and evidence in relation to Post-Accident accessibility to ensure the plant can be returned to a safe, stable state and ensure exposures are controlled below statutory limits where applicable and demonstrate doses are reduced so far as is reasonably practicable.

346. Hitachi-GEs strategy is to identify bounding faults within the Design Basis and those which are beyond Design Basis and within the Severe Accident definition. These are

then used to identify the relevant Structures, Systems and Components which would require manual intervention to allow the plant to be stabilised.

347. For the purpose of providing a meaningful assessment, Hitachi-GE identifies the most realistic route (shortest) and a reasonably conservative (longer) route to be used by operators from the MCR to the relevant SSC to be used in the mitigative action, and identify the duration and potential exposures based on calculated dose rates to carry out those actions.

348. Hitachi-GE also provides representative examples of exposures, both internal and external, to be expected by MCR operators during the period of the events being postulated. Hitachi-GE defines the term post-accident to be “the period from the initiation of a postulated accident to the period when the plant is returned to a stable condition (typically a few hours to several days)”. They specifically exclude recovery operations e.g. recovery of fuel or material as these are seen as recovery/remediation operations.

4.2.8.1 Design Basis Accident

349. Hitachi-GE identifies a Design Basis Accident of a double ended guillotine rupture of a limiting line within the primary containment. After an initial blowdown into the primary containment (and the resultant build-up of pressure), radioactive material is identified as escaping to the environment via two routes:

- From primary to secondary containment where it is captured via the filtered stand-by gas treatment system (SGTS) and released via the stack.
- From the primary to secondary containment through the main steam isolation valves (MSIVs) and the main steam line (MSL) to the Turbine Building (T/B), with a subsequent release to the environment at ground level.

350. Hitachi-GE identify the representative example of an SSC requiring intervention as the Light Oil Tank (LOT) within the Emergency Diesel Generator Building (EDG/B) of which there are 3 LOTs and EDG/Bs, only one of which is required to operate in order to support the function required. The LOT supporting the EDG/B on load during the postulated event will require refilling within 7 days as it has a 7 day capacity and if it is not replenished then emergency cooling will be lost with the loss of electrical power.

4.2.8.2 Beyond Design Basis and Severe Accident

351. In the case of Beyond Design Based Accidents and Severe Accidents Hitachi-GE identifies that management of the event can be supported through the MCR or the Back-Up Building (B/B) in cases where PCV failure is postulated. Hitachi-GE assumes operators would relocate from the MCR to the B/B control room before failure of the PCV and so any exposure assessment for this transfer is not relevant. Since there would seem to be sufficient time and warning for operators to relocate to the B/B this would seem to be a reasonable assumption within GDA. The Severe Accident sequence identified as a representative example is a (Non-LOCA event with failure of control rod insertion and core cooling, resulting in high pressure core damage in the short term) this provides the most severe environmental conditions of all the SA sequences in which the Flooder System of Reactor Building (FLSR) is credited. The FLSR is a manually operated system that needs to be directly accessed by the operator.

352. Hitachi-GE identifies in the PCSR that if noble gases should enter the MCR exposures to the MCR operators could exceed relevant reference dose levels. In response to this Hitachi-GE considered the use of a shielded shelter beside the MCR, which is the option identified in the reference plant. Hitachi-GE notes the shielded shelter is not the only option available and any hardening to the existing design must be done in line

with the accident management philosophy, which determines those conditions where habitability of the MCR must be maintained and those when evacuation to an alternative control centre must be carried out. Hitachi-GE determines these changes should be reviewed at site specific phase. I believe this to be a reasonable position to take at this time based on the work done to date and the demonstration of activities, actions and exposures calculated.

4.2.8.3 Exposures to MCR and Intervention workers during DBA Events

353. The total dose, internal and external components, to which the MCR worker is estimated to be exposed during this postulated event is 5.9E-09 Sv (5.9E-6 mSv). No time has been accounted for carrying out the intervention tasks of refuelling the LOT.

354. The total dose assessed for the actions related to refuelling the LOT has been calculated as 6.2E-08 Sv (6.2E-05 mSv), received whilst traversing the T/B, and 1.3E-10 Sv (1.3E-07 mSv) whilst traversing the C/B, the outside areas between T/B and EDG/B and within the EDG/B whilst carrying out the refuelling. This would give a cumulative exposure of 6.8E-08 Sv (6.8E-05 mSv).

355. If the event was to continue over a 30 day period and assuming the LOT requires filling every 144 hours then a maximum of 5 interventions would be required to ensure supply was uninterrupted. Hitachi-GE extrapolate dose rate data from dose profiles calculated in relation to exposures over time within the T/B and those external to the building and apply these to each of the 5 interventions. Hitachi-GE calculate a total exposure of the 5 interventions of 4.3E-07 Sv (4.3E-04 mSv) or 0.43 µSv. It should be noted that the total exposure is not simply a summation of 5 interventions using the cumulative exposure detailed in paragraph 354 as it is expected dose rates within the T/B will increase until stabilising at around 200 hours after the initiating event. External exposures remain relatively stable rising slightly to the end of the 30 day period.

356. Chapter 30 (Ref. 110) details the number of personnel and nature of the roles required to operate the unit and station during all modes of operation into both DBA and SA events. The MCR operations will be supported by a minimum of two SQEP operators at all times with a Control Room Operator (CRO) and a supervisor.

4.2.8.4 Exposures to MCR and Intervention Workers during a Beyond Design Basis and Severe Accident Event

357. A 12 hour shift or working time is assumed for the workers within the MCR during the SA assessment. This leads to an external dose from the ventilation system due to intake of contaminated air from the outside. The dose for those within the MCR during the 12 hour period is 4.7E+03 mSv worst case, based on initial assumptions, and prior to mitigation / dose reduction measures: This is significant.

358. Workers attending to the SSCs outside of the MCR receive an effective dose from both external and internal exposures of 8.2E-02 mSv

359. Application of exposure reduction measures with respect to MCR doses primarily from direct radiation dose from external air within the HVAC system provides a revised exposure to 3.3E+01 mSv or 33 mSv and a total dose for a worker carrying out intervention works outside and inside the MCR for a total period of 13 hours is 5.0E+01 mSv or 50 mSv.

360. RQ-ABWR-1367 (Ref. 111) was raised to clarify a number of points to ensure the assessment is bounding and no other fault scenarios or assumptions needed to be brought into the assessment. Hitachi-GE confirmed an intervention using the FLSR would be carried out externally to the building 8 hours from the event initiation. The dose rate is estimated to be 8.2E-02 mSv/h. Thus a 1 hour task is estimated to be

0.082 mSv or approximately 0.1 mSv. Hitachi-GE then make a comparison later in the venting cycle where the dose rate external to the buildings is 6 mSv/h. This would lead to a dose of 6 mSv. Hitachi-GE compares this dose against a constraint of 100 mSv based on REPPIR. Following Hitachi-GEs response to queries and clarification of assumptions I consider this to be reasonable at this stage in GDA.

4.2.8.5 Emergency Facilities

361. Chapter 22 (Ref. 112) of the PCSR describes the overview of emergency facilities to support the UK ABWR Generic Design in case of events leading to accidents and emergencies even during the most severe event environments. The facilities described will include both on and off-site support. Hitachi-GE states that further to these facilities the emergency response will be supplemented by alternative facilities or arrangements to ensure a flexible and resilient response if the main facilities become untenable or are unsuitable to manage the event scenario.

362. Hitachi-GE identifies dedicated emergency response facilities within other operational buildings on-site.

363. The dedicated facilities consist of the Emergency Control Centre (ECC) which is reserved for use when required by trained responders led by the Emergency Controller who has strategic control of the on-site response. The ECCs functions will include:

- Formulation of public countermeasure advice.
- Collection, co-ordination and dissemination of incident information.
- Environmental and radiological information as well as mustering of staff and setting the site response aims, focuses and actions.
- Provision of technical advice to the control room and other emergency centres and formulation of media statements and focused media response.

364. There is a Technical Support Centre adjacent to the ECC which will be populated by the Technical Support Team. The team will have access to plant data in the form of live reactor telemetry and data displays, manuals, drawings and emergency support processes (Emergency Operation Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs)). They provide advice and direction for the control room team to assist in controlling the reactor or plant event.

365. The PCSR provides an overview of how the on-site facilities interact and communicate with off-site facilities and agencies identified to provide support to the event.

366. Hitachi-GE identifies the need for emergency arrangements to be accepted by the regulator prior to first nuclear fuel being received onto site, although it does not identify which facilities will need to have been constructed and commissioned prior to this event.

4.3 Regulatory Issues

367. Regulatory Issues (RIs) are matters that ONR judge to represent a 'significant safety shortfall' in the safety case or design and are the most serious regulatory concerns. RIs are required to be addressed before a DAC can be issued.

368. Although no Regulatory Issues were raised directly by the Radiological Protection Specialism, one of particular relevance was raised by Reactor Chemistry, RI-ABWR-0001: Definition and Justification for the Radioactive Source Terms in the UK ABWR during Normal Operations (Ref. 113) raised on the 02nd June 2015 and which was closed by Hitachi-GE as agreed by ONR on 19th October 2016 (Ref.114).

369. This issue was primarily raised to focus Hitachi-GEs programme relating to definition of the source term provided to support assessment of the UK ABWR into a format and

with the required level of detail to meet the UK regulatory requirements as expected by ONR and EA.

370. It should be noted that Hitachi-GE generally operates within a prescriptive regulatory environment where specific conditions and parameters are placed around most elements of safety case development and assessment. The UKs goal setting regime, although containing certain targets and limits places minimal constraint or direction on the development of the case.

371. The original submission in relation to the source term provided an unduly large selection of nuclide data, with little acknowledgement of the significance of the nuclides being presented or what the data was being provided for and in what context. Reactor Chemistry and the EA took a lead role in raising the issue and tracking this through to the successful resolution. Radiological Protection had appropriate input in relation to the development of an appropriate source term with respect to the End User where it impacted radiological protection assessments of exposures to workers and the public, shielding and releases.

372. The general progress and development of this issue with respect to Radiological Protection is primarily discussed earlier in this report within section 4.2.1 Assessment Findings and minor shortfalls arising from the assessment with respect to Radiological Protection are discussed within this earlier section.

4.4 Regulatory Observations

373. Regulatory Observations (ROs) are raised when ONR identifies a potential regulatory shortfall which requires action and new work by Hitachi-GE for it to be resolved. Each RO can have several associated actions.

374. There were three RO’s where Radiation Protection was the main technical lead. These shall be discussed within this section.

- RO-ABWR-0014.
- RO-ABWR-0064.
- RO-ABWR-0065.

375. There were 26 RO’s where Radiation Protection was a related technical area; these have been either discussed within this Step 4 assessment report or have been resolved by other specialisms.

376. A summary of ROs where radiation protection was the main technical lead can be found in Annex 4.

4.4.1 RO-ABWR-0014

377. During the early stages of the GDA I set expectations of what is required for a radiological protection safety case. Information provided by Hitachi-GE did not meet expectations and so an RO was raised (Ref. 115) for Hitachi-GE to provide UK ABWR Radiological Protection Safety case: project plan and delivery.

378. Four RO actions were raised:

- Provide a strategy for development of the radiological protection elements of the UK ABWR safety case.
- Provide a project plan for delivery of the radiological protection elements of the UK ABWR safety case.
- Allocate suitably qualified and experienced resources to develop the UK ABWR radiological protection safety case.
- Radiological protection safety case deliverables.

379. For the above actions Hitachi-GE responded by providing a resolution plan (Ref. 116). Within the resolution plan it stated that one document (Ref. 117) would be submitted to close the above actions.

380. For the first action, information was provided (Ref. 117) which details a step by step strategy for the development of the radiological protection safety case. This included the Hitachi-GE strategy in understanding the ALARP philosophy.

381. For the second action the document (Ref. 117) identified the various reports that shall formed part of the radiological protection safety case. This was to show that a claim, argument and evidence based approach was to be undertaken. A chart of when these reports would be delivered was also provided.

382. For the third action the initial information provided (Ref. 117) was unsatisfactory and additional information was requested through an RQ (Ref. 89). In response to the RQ Hitachi-GE provided information on role profiles, and CV's of Hitachi-GE personnel involved within the GDA from an radiation protection perspective. This information was appropriate to close the action.

383. For the fourth action a chart was provided detailing the initial timescale for the deliverables of radiological protection safety case. It should be noted that all documents have been submitted at the point of writing this report.

384. From assessing the information provided I was satisfied that all of the actions for the RO could be closed in June 2016 (Ref. 118). No minor shortfalls or assessment findings are related to the closure of this RO.

4.4.2 RO-ABWR-0064

385. From initial discussions with, and review of documentation supplied in Step 3 by Hitachi-GE it was not possible to identify clearly the approach that was to be undertaken in controlling radioactive contamination. It is expected that the UK ABWR is designed such that permanent and temporary features required to manage and prevent the spread of radioactive contamination, from areas of high designation to those of lower designation are fully considered. Although an initial RQ was raised (Ref. 119), the response provided a high level statement on the design philosophy used. It did not address to an appropriate level of detail the specifications required for surface preparation and examples of locations of features to which contamination control is applied. An RO was raised (Ref. 120) in October 2015 to address contamination control.

386. In May 2016 additional objectives were added to the RO regarding the identification of appropriate controls including interlocks to provide protection against unplanned, unexpected exposures including overexposure (Ref. 121).

387. This RO is cross cutting and was raised to ensure the design of the UK ABWR includes appropriate arrangements for both permanent and temporary features necessary for the adequate control of contamination and for the prevention of over-exposure to radiation. It was also a requirement to ensure these features were maintained through all phases and stages of operation of the UK ABWR.

388. Eleven RO actions were raised:

- Action One: Hitachi-GE provide a Resolution Plan.
- Action Two: Identify and present the locations, nature and extent of potential radioactive contamination.
- Action Three: Explain the design philosophy in relation to the control and containment of radioactive material. This should include all aspects of

containment control through fixed features, including ventilation and barriers, as well as through the provision of moveable features.

- Action FourIdentify and present the relevant standards from which specifications for materials, surfaces and surface-finishes are identified in relation to minimising contamination adherence, and to aid with decontamination.
- Action Five: Following the identification of the specifications which define the materials, surfaces and surface-finishes used in areas with the potential to become radioactively contaminated provide examples of how they are applied within the UK ABWR design.
- Action Six: Identify the features needed by the design to facilitate decontamination techniques prior to intrusive maintenance or following an unplanned leak from primary containment.
- Action Seven: Identify how Hitachi-GE intends to manage HVAC/LEV arrangements within the relevant buildings of the UK ABWR design to ensure a balanced and controlled cascade ventilation system is maintained.
- Action Eight: Identify the nature and location of monitoring (airborne, radiation and surface contamination) equipment required to control and minimise contamination spread within the UK ABWR, providing examples.
- Action Nine: RP to provide the strategy by which access to high dose rate and high dose areas will be controlled by physical means such as interlocks, alarms, or locked doors to prevent unauthorised entry.
- Action Ten: Following on from the above action can Hitachi-GE provide details of the categorisation and classification of the relevant interlocks or engineered protective systems in line with Hitachi-GE Safety Case manual.
- Action Eleven: Can Hitachi-GE identify the necessary EMIT arrangements for the identified interlocks or engineered protective systems as identified above.

389. It should be noted that a significant number of RQs were raised to support the closure of RO-ABWR-0064. Only the most relevant RQs are discussed within this section.

4.4.2.1 Action One

390. Hitachi-GE provided several revisions of the resolution plan; the latest one (Ref. 122) provided information on the documentation that was used to resolve the RO actions identified above.

- Locations, Nature and Extent in relation to Radiation and Contamination (Ref. 66).
- Contamination Control Philosophy (Ref. 123).
- Topic Report on Radiation and Contamination Monitoring of Occupational Exposure (Ref. 124).
- UK ABWR Design Strategy for Access Control to High Dose Rate and High Dose Areas (Ref.99).
- Access Control to High Dose Rate and High Dose Areas on UK ABWR: Representative Examples (Ref. 125).
- Contamination Control and Protection against Direct radiation: Design Study for UK ABWR (Ref. 54).

The resolution plan provided appropriate information; milestones and timescales proposed were acceptable so allowing closure of the action.

391. It should be noted that a significant number of RQs were raised to support the closure of RO-ABWR-0064. Only the most relevant RQs are discussed within this section.

4.4.2.2 Action Two

392. Hitachi-GE provided appropriate information for action two in the paper 'Locations, Nature and Extent in relation to Radiation and Contamination' (Ref. 66). The document provided appropriate worker activities, providing a high level review of the activity as well as the potential nature of contamination (whether solid, liquid or gas). Information on these worker activities can be found in section 4.2.3.5. Overall the information is adequate.

4.4.2.3 Action Three

393. Hitachi-GE provided information for action three in the paper 'Contamination Control Philosophy' (Ref. 123). The document provides a review of RGP within both international and the UK Nuclear Industry. The report included aspects of contamination control through fixed features (i.e. barriers / ventilation). The document did not, however succinctly detail an overarching philosophy statement of how Hitachi-GE approached contamination control. Information was split between the submissions and so lacked a level of clarity. It will be necessary for the Licensee to develop an overall succinct philosophy regarding design for contamination control; this has been raised as an Assessment Finding (AF-ABWR-RP-04).

AF-ABWR-RP-04: *The Licensee shall develop an overarching philosophy regarding design for contamination control for the UK ABWR. This should ensure adequate source minimisation and control through the use of engineered features to minimise chronic and acute leakage. This should include but not be limited to use of HVAC and LEV, appropriate detection, and controls such as bunding and drainage.*

4.4.2.4 Action Four

394. For action four relevant standards and guidance are supplied within the Contamination Control Philosophy (Ref. 123). Hitachi-GE uses its Nuclear Safety and Environmental Design Principles (NSEDPs) along with the standards/guidelines to create Mitigating Options (MOs) which follow the ERIC-PD methodology. These MOs then stipulate how to eliminate, prevent or stop contamination spread for the appropriate case studies. The information provided is considered appropriate for GDA and so this action is considered closed.

4.4.2.5 Action Five

395. For action five, case studies in Contamination Control Philosophy (Ref. 123) provide evidence Hitachi-GE has an appropriate understanding of RGP to control and reduce contamination levels for each aspect of the UK ABWR life cycle. As stated previously Hitachi-GE uses its NSEDPs along with the standards/guidelines to create Mitigating Options (MOs). This follows the ERIC-PD methodology. Relevant detailed information for each case study is available, though in a small number of cases the MOs provide limited evidence to support the associated claim and argument. This is apparent for material selection (a potential outcome from this is an effect on the amount of cobalt that builds up within components which in turn increases the radiation and contamination risk to workers).

396. RO-ABWR-0035 (Robust Justification for the Materials selected for UK ABWR) contains a number of actions; two actions (Ref. 126) are related to action five of RO-ABWR-0064 examples of material selection.

- Hitachi-GE to provide a robust justification that the amount of high cobalt alloy usage in UK ABWR has been reduced SFAIRP.
- Hitachi-GE to provide a robust justification that the treatments applied to material surfaces in UK ABWR reduce radioactivity SFAIRP.

397. Information regarding RO-ABWR-0035 and the associated actions which provide further detail beyond RO-ABWR-0064 can be obtained from the reactor chemistry assessment of the UK ABWR (Ref. 38). In summary the actions were successfully closed.

4.4.2.6 Action Six

398. For action six, information is provided within Contamination Control and Protection against Direct Radiation: Design Study for UK ABWR (Ref. 54). Initial Hitachi-GE documentation provides a high level breakdown of the MOs for each of the specific worker activities discussed within section 4.2.7.3. The MOs detail techniques to be used to help control and mitigate the spread of contamination and to help with the decontamination during maintenance.

399. As the information provided was at a high level, a workshop was held at the beginning of 2017 to review in greater detail two of the eight high worker dose activities. The output of the workshop was several RQs to clarify details discussed.

400. The response to RQ-ABWR-1311 (Ref. 127) provides information on the methodology for FMCRD maintenance. One aspect to be discussed was the approach to be undertaken to drain cooling water from the machine during maintenance. For the Lower Component (L/C) and Upper Component (U/C) appropriate draining techniques are employed to facilitate source reduction, decontamination and therefore reduce the spread of contamination within the lower drywell. However the CRD handling device is designed such that leaking water flows out of the machine and is guided through a tube to the floor of the lower drywell before being drained into a local contaminated water drain. I do not consider this to be an ALARP measure as within the SAPs (Ref. 6) stipulates that levels of contamination should be kept ALARP, taking into account the nature of the activities being undertaken. It is potentially feasible for an engineered solution to channel any leak direct to a sealed drain or even a storage tank to reduce spread of contamination. A similar case is apparent where the L/C is drained into a spool piece work table. In this case there is a potential for spillage or aerosol generation as the L/C is drained by hand into a large Tundish style drain. As the control of contamination at source is one of the key Radiological Protection SAPs (RP.4) (Ref. 6) an assessment finding (AF-ABWR-RP-04) has been placed regarding the appropriate control and management of containment regarding minor chronic and acute leaks/spillages.

401. The remaining high dose worker activity reviewed was the opening and closing of the RPV head where several RQ's (RQ-ABWR-1065 (Ref. 128), RQ-ABWR-1312 (Ref. 129)) were raised. This is one of the main worker activities which involve substantial contamination control due to the complexity of the task. Within the RQ responses there is a breakdown of each task as well as appropriate information regarding containment measures (e.g. sub change facilities, temporary C3 containment) to control the spread of contamination as well as decontamination techniques. One of the decontamination techniques is polishing of the seal surface before putting a protective layer over the seal surface before the reactor is flooded. The same process is undertaken when installing the RPV head but in reverse. From reviewing the information, it was unclear why the reactor seal has to be polished twice and I questioned whether it would be more appropriate to polish the seal once when installing the RPV lid. This would in effect reduce the amount of time operators need to be in the RPV well when the lid has been removed as well as reduce the amount of waste produced.

402. A similar situation is present for Reactor Well Decontamination (another of the eight high dose worker activities) where the following steps are undertaken.

- Cover sheets are put on the bottom half of the reactor to prevent radioactive cruds on the reactor well. It should be noted that only the bottom half has protective covering the top half has no covering.
- The top half of the reactor is decontaminated using an automated device.
- For the bottom half of the reactor well, once the water has been drained operators decontaminate the cover sheets before removing them.
- Operators then decontaminate the bottom half of the RPV well.

This raises the question what are the advantages of using the cover sheets if the bottom reactor well is still required to be decontaminated. There appears to be an opportunity to simplify the decontamination procedure so as to reduce operator dose and active waste produced.

403. From the information provided, there appears to be an opportunity to reduce duplicated tasks in order to reduce worker doses and the generation of radioactive waste. An assessment finding for future licensees has been raised regarding this matter

AF-ABWR-RP-05: *The Licensee shall review the design for opportunities to minimise duplicate tasks within work programmes required for decontamination tasks taking due account of potential detriments (worker dose, secondary waste creation) to ensure risks are controlled so far as is reasonably practicable.*

404. In summary the evidence that has been provided for action six of RO-ABWR-0064 at a high level provided appropriate information with how the UK ABWR has been design to facilitate decontamination techniques prior to intrusive maintenance or an unplanned leak. However from a detailed review of couple of examples further work is required. Assessment finding AF-ABWR-RP-05 was raised regarding this action.

4.4.2.7 Action Seven

405. Evidence for action 7 of RO-ABWR-0064 was provided by Hitachi-GE across a number of documents. The primary documents supplied within the response were Contamination Control Philosophy (Ref. 123), which provided an overall statement of approach and a series of good practice referenced within case studies. It should be noted the document did not necessarily detail a philosophy but did provide reassurance of the design intent and provided appropriate references and case studies for which ONR was able to assess the approaches to be adopted. The second document provided was the Contamination Control and Protection against Direct Radiation: Design Study for UK ABWR (Ref. 54). This provided the application of the case study examples against the design of the UK ABWR to demonstrate the use of international good practice.

406. Further to this as part of the cross cutting work, ONR Radiological Protection Inspectors reviewed submissions in response to RO-ABWR-0075 “Robust demonstration that the design of the UK ABWR HVAC system has been adequately conceived and reduces risk SFAIRP” (Ref. 130) related to HVAC systems specified for the UKABWR. This review also provided evidence to support closure of Action 7 of RO-ABWR-0064 as detailed below. Further information on the cross cutting assessment of HVAC systems is detailed in section 4.5.2.

407. Documentation presented demonstrated evidence in relation to the application of cascade ventilation designed to provide adequate depressions leading to appropriate flow rates and room air changes in a required period. Consideration of the need to balance the HVAC system and account for changes in external pressures along with the need to facilitate access to areas by ensuring doors can be opened against potential differential pressures between areas are also presented. There is evidence of due consideration for the need to access and maintain components along with the need to change HEPA filters and manage such filters.

408. There is however limited evidence of consideration for the need to provide adequate airborne monitoring within the HVAC and filter bank areas along with area radiation monitoring to provide early warnings of loss of containment and prevent access to areas of potential elevated contamination and radiation. This is seen as an assessment finding (AF-ABWR-RP-06).

409. In general the response to RQ-ABWR-0064 Action 7 meets its intent with Hitachi-GE presenting information on how it intends to manage HVAC and to an extent LEV within the relevant buildings of the GDA design to ensure a balanced and controlled cascade ventilation system. There are a number of assessment findings which are cross cutting in relation to the system and specifically relating to other actions within RO-ABWR-0064.

AF-ABWR-RP-06: *The Licensee shall develop an assessment on the use of LEV based around the specific hazards posed and develop the necessary controls and arrangements. This is required as the implementation of fixed engineering controls should take precedence over mobile units due to the inherent nature of management and use of such units.*

4.4.2.8 Action Eight

410. For action eight, Hitachi-GE provided ‘Topic Report on Radiation and Contamination Monitoring of Occupational Exposure’ (Ref. 96). This provides the majority of information relating to the type, location and equipment to be used to monitor contamination and restrict any potential migration from areas of higher classification to lower classification. It should be noted that at the time of reviewing the response to action eight, revision 3 of Radiation and Contamination Monitoring of Occupational Exposure (Ref. 96) was not available and the assessment is based on revision 2 of the topic report (Ref. 124). It was noted that in a recent RQ response (Ref. 132) regarding the radiation monitoring alarm system within the Off-Gas system, that no local alarms were installed to warn operators in the region of a potential release of radioactive material or high dose rate. The main control room would be responsible for initiating the alarm procedure. Following the hierarchy of control measures, it would be prudent to install local alarms to warn operators of the necessary action required immediately so as to reduce the risk of unintended consequence; this concern is to be taken forward under assessment finding (AF-UKABWR-03) section 4.2.5.3. It should be noted that this assessment finding is also linked to RO-ABWR-0073 (Robust demonstration that the design of the UK ABWR off-gas system reduces risks SFAIRP).

4.4.2.9 Actions. Nine, Ten and Eleven

411. The last three actions relate to access arrangements and interlocks. The SAPs (Ref. 6) RP.7 stipulate that a hierarchy of control measures should be implemented to optimise protection. This is based on IRR99 (Ref.14) reg. 8(2) where an engineered approach should be taken to restrict exposure SFAIRP; only when this is not feasible should a safe system of work be implemented. From an initial review of RP access arrangements it appears that a safe system of work was to be applied instead of an engineered approach. With this in mind an action nine, ten and eleven were added to RO-ABWR-0064 to capture this shortfall.

412. Hitachi-GE provided two main documents ‘UK ABWR Design Strategy for Access Control to High Dose Rate and High Dose Areas’ (Ref. 99) and ‘Access Control to High Dose Rate and High Dose Areas on UK ABWR: Representative Examples’ (Ref. 125) to respond to these actions.

413. The design strategy document (Ref. 99) provides appropriate information regarding Hitachi-GE’s approach for access control to high dose rate and high dose areas for the UK ABWR. It outlines an approach whereby Hitachi-GE estimates the radiological risks

and reviews the information against the UK ABWR category and classification. Once the classification has been identified Hitachi-GE will review the different options for access control either by implementing an engineered approach (interlocks) or by a safe system of work. The final part of the document provides information on the appropriate level of EMIT that shall be used to maintain access arrangements. Overall this document is adequate from a radiological protection aspect.

414. A representative example (Ref. 125) is provided as evidence that the design strategy to be undertaken (Ref. 99) is appropriate. Although the example provides the appropriate information required to corroborate the design strategy (Ref. 99) only one example was provided. Several further RQ's were raised (i.e. RQ-ABWR-1395 (Ref. 73), RQ-ABWR-1397(Ref. 131) and RQ-ABWR-1417(Ref. 132)) to provide further examples as evidence of access control using different interlocking mechanisms being implemented within the UK ABWR design. Hitachi-GE responded by stating that the future licensee will review and adopt the design strategy. Consequently this concern is taken forward under assessment finding (AF-ABWR-RP-03) section 4.2.5.3

415. As RO-ABWR-0064 is cross cutting with various disciplines further assessment findings and minor shortfalls have been raised; these are discussed in respective disciplines' assessment reports.

416. I was broadly satisfied with the information provided and closed the RO in July 2017 (Ref. 133). Several assessment findings are related to the closure of this RO as well as information within RO-ABWR-0035.

4.4.3 RO-ABWR-0065

417. During cross cutting discussions involving myself, the LMR Inspector and Hitachi-GE concerns were raised regarding the general use of lead wool as a flexible shielding material within the generic design. Lead wool had been identified as the primary material to make up any gaps between services and shielding walls within through-wall penetrations.

418. Lead can cause serious health problems such as anaemia or kidney disease and published research has linked exposure to a small number of occupational cancers. Its use is controlled under The Control of Lead at Work (CLAW) Regulations 2002 (Ref. 134). Lead is also difficult to dispose of and will present issues during maintenance, refurbishment and decommissioning. Lead wool presents a further challenge as its form increases the generation of lead dust in the working atmosphere when handled during construction or during maintenance or decommissioning. The (CLAW) Regulations (Ref. 134) use the hierarchy of controls as a basis and as such alternative materials should be sought where practicable.

419. To gain further information on this subject an RQ was raised (Ref. 135). The response along with further discussions identified the extensive use of lead wool within the design. I along with the LMR Inspector and the EA assessed the response with the arguments and evidence as being below expectations, taking into account modern designs and availability of alternate materials. To this extent RO-ABWR-0065 was raised to seek appropriate resolution to the use of Lead Wool and to the design of suitable penetrations using inherently safe techniques and without the inherent reliance on lead wool so as to minimise the use of lead within the design.

420. Following the issue of RO-ABWR-0065 Hitachi-GE undertook a review of its Penetration Design Rules and its overall process along with a review of the nature of Penetrations within the design. They also undertook a review of alternative materials which could be used within the design to substitute lead wool.

421. Seven RO actions were raised:

- Action One: Hitachi-GE to provide a resolution plan detailing the process to be followed, and how it intends to comply with the remaining actions.
- Action Two: Hitachi-GE to identify the number, location and configuration of penetrations though shielding structures within the UK ABWR generic design.
- Action Three: Hitachi-GE will identify the nature (i.e. Radiation, type(s)) and level of hazard posed by each penetration from the relevant radiation sources.
- Action Four: Hitachi-GE to identify any potential competing requirements in relation to the penetrations identified.
- Action Five: Hitachi-GE to identify a range of solutions which could be applied to the identified penetrations based on the results of the previous action.
- Action Six: Hitachi-GE to report to ONR on the output of the previous actions.
- Action Seven: Hitachi-GE to revise all relevant documentation including the PCSR accordingly to reflect the output of this RO.

4.4.3.1 Action One

422. Hitachi-GE provided the resolution plan (Ref. 136) for review; within the plan it stipulated that two documents would be submitted to resolve the above RO actions.

- Penetration Design Guideline (Ref. 137).
- UK ABWR Penetration Design Rule (Ref. 86).

In addition to these documents an RQ was raised (Ref. 138) to resolve outstanding queries relating to the RO. The Resolution plan was fit for purpose and this action was closed.

4.4.3.2 Action Two

423. Hitachi-GE provided a table within UK ABWR Penetration Design Rule (Ref. 86) detailing the number and location of penetrations through shielding structures across the reference design for the UK ABWR (latest Japanese ABWR). Further clarification was provided with RQ-ABWR-1313 (Ref. 138) regarding pipework and penetrations; the information provided was adequate at a GDA level to close the action.

4.4.3.3 Action Three

424. Within UK ABWR Penetration Design Rule (Ref. 86), Hitachi-GE provided examples of the approach that shall be undertaken by the future licensee when reviewing penetrations. One of the initial steps Hitachi-GE undertook was to understand the Radiation shielding requirements. This involves looking at radiation source types as well as the radiological zoning and worker occupancy for penetration design. Hitachi-GE presented information to allow this action to be closed.

4.4.3.4 Action Four

425. Penetration Design Guideline (Ref. 137) provides a breakdown of the design requirements that must be reviewed for each type of penetration within the UK ABWR.

- Radiological Shielding.
- Containment.
- Pressure (Internal/process).
- Sealing Pressure (across boundary).
- Temperature.
- Structural.
- Fire Resistant.
- SSC Requirements.
- Material Compatibility.
- Service Life.

- Segregation of systems.
- Maintenance Requirements.
- Accessibility.
- Decommissioning.

Appropriate examples are provided in both UK ABWR Penetration Design Rule (Ref. 86) and RQ-ABWR-1313 (Ref. 138) when reviewing each requirement for a particular penetration design. This action is closed.

4.4.3.5 Action Five

426. Penetration Design Guideline (Ref. 137) provides different penetration options for indirect penetration (‘joggling’), straight through penetration or penetrating with compensating shielding. The information provided is quite detailed and provides options when taking into account the above design requirement options. This action is closed.

4.4.3.6 Action Six

427. Within UK ABWR Penetration Design Rule (Ref. 86) several examples are provided of penetrations within the UK ABWR reference design. For each example the design requirement options previously stipulated are assessed to find the ALARP option. However for each of the examples chosen the first option ‘straight through’ penetration was the ALARP option and joggling or with compensating shielding. Within RQ-ABWR-1313 (Ref. 138) Hitachi-GE provided an appropriate example where indirect penetration was the best ALARP option. Although this has satisfied the specific action for the RO an assessment finding was raised (AF-ABWR-RP-07) see action Seven (4.4.3.7). As this is at a GDA stage and not site specific it will be up to the future licensee to follow the Hitachi-GE approach for UK ABWR Penetration Design as well as make sure that the ALARP option is undertaken.

4.4.3.7 Action Seven

428. Within the latest submission of the PCSR (Ref. 31) information regarding penetrations is provided within chapter 20.

429. It should be noted that within Penetration Design Guideline (Ref. 137) the appendices review by Hitachi-GE for alternative shielding materials (to lead) for the penetration pipes. At the time of writing this report there is no clear preferred option stipulated and so this shall be reviewed at site license stage; this has led to an assessment finding AF-ABWR-RP-07.

AF-ABWR-RP-07: *The licensee shall develop penetration design in line with the proposed philosophy and design rule, applying inherently safe designs for penetrations ensuring use of lead wool is minimised or where reasonably practicable removed to ensure risks to employees are reduced so far as is reasonably practicable.*

430. As RO-ABWR-0065 is cross cutting with various disciplines further assessment findings and minor shortfalls have been raised; these are discussed in respective disciplines’ assessment reports.

431. I was broadly satisfied with the information provided and closed the RO in May 2017 (Ref. 139).

4.5 Cross-Cutting

4.5.1 Bottom Drain Line

432. During Step 2 and Step 3 of GDA, ONR identified a feature within the UKABWR design known as the BDL. The BDL is a feature which directs and contains a flow of reactor coolant between the RPV and the CUW.

433. The BDL emerges from the base of the RPV as a 73 mm external diameter pipe. It traverses through and emerges from the CRD housings before traversing the lower dry well area. This became of interest to Reactor Chemistry, Structural Integrity, Fault Studies and Radiological Protection due to the potential for degradation and potential LOCA events and also with regard to radiological exposures not only during any such LOCA but also due to irradiation workers from the build-up of ^{60}Co inner surface corrosion films of the BDL.

434. Workers have to enter the lower dry well during outages and as such the BDL will be present as a source of radiation. It will also require EMIT to demonstrate its continued integrity is maintained and will lead to further exposures in carrying out this work. Therefore the benefits of its retention within the design need to outweigh the hazards and associated risks posed by the interventions required to maintain it and its impact on other operations where workers are exposed.

435. The BDL is referenced in the Reactor Chemistry and Structural Integrity Topic Reports and so a summary is provided here with respect to the radiological protection elements.

436. The BDL is present within the Drywell at all times and the design of the BDL and Drywell means that workers would be in close proximity to the BDL during routine outages and maintenance operations being carried out during those outages. Hitachi-GE estimates a fivefold reduction in dose rates emanating from the BDL in the UK ABWR this being less than 5 mSv/h at contact and less than 0.5 mSv/h at 1 meter (shielding),

437. These concerns led to the development of RO-ABWR-0034 “Demonstrating the Inclusion of a “Bottom Drain Line” in the UK ABWR Design achieves inherent safety and reduces risks SFAIRP” (Ref. 140).

438. The TSC reviewed German and international OPEX relating to BDL (Ref. 141). German designs either removed and plugged the BDL or were designed not to include the feature. The German OPEX review could not reveal any compelling evidence for its inclusion in the design. This demonstrates BWRs can operate without this feature meaning that Hitachi-GE should have a compelling argument to retain it. Hitachi-GE’s evidence of the development of the BDL within the BWR design points to its use being to aid in removal of CRUD. As the design has developed CRUD has become less of an issue, although build-up of deposits within reactor coolant circuits is undesirable as it reduces efficiency and creates hot spots of radioactive material creating dose issues for maintenance workers.

439. Hitachi-GE identified a number of reasons to retain the BDL within the design. These included: greater control of the thermal stratification within the RPV through better monitoring of the differential temperatures through inclusion of a temperature measurement feature in the BDL, which if the BDL were removed would not be available; monitoring and maintaining primary coolant water quality; monitoring water level below the fuels in the RPV in severe accident conditions; and mitigating the risk of fuel failure by removal of potential debris.

440. During the period of inquiry into the BDL, its design, construction and justification for inclusion based on overall risk, a series of RQs were raised across the major specialist areas of Reactor Chemistry, Structural Integrity, Radiological Protection and Conventional Health and Safety and Human Factors. Hitachi-GE responded to all

queries and over the period and has provided sufficient argument and evidence to meet the needs of GDA.

441. The primary focus for Radiological Protection is to ensure exposures are restricted SFAIRP. Workers are routinely exposed during reactor plant outages, where examination, inspection, maintenance and testing is carried out on the: RPV, BDL, FMCRD, RIP, and other systems which require EMIT under the plants EMIT schedule. Average dose rates from the Reference Plant KK-7, taken over representative outages are up to 1 mSv/h at 1metre and less than 20 mSv/h at contact with the BDL, both are without the local insulator/shielding. The estimate of dose rate at 1 metre with the insulator shield in place is approximately 1/10th of the dose without the insulator in place. The doses associated with BDL inspection from the reference plant are up to approximately 4 person-mSv. The BDL also contributes to collective dose for RIP maintenance, with exposures for RIP maintenance less than 1 person-mSv average and a bounding case less than 4 person-mSv per outage.

442. Hitachi-GE has made a number of modifications to the design which include: reactor chemistry applying HWC, OLNC and DZO, provision of remote inspection for large proportions of the BDL as part of the EMIT arrangements, consideration of alternative materials to aid in dose reduction, provision of improved data, and assessment of the source terms based on materials selection. Hitachi-GE has estimated this will lead to a dose rate reduction in the order of 80% from dose rates experienced at the reference plant KK-7. Dose rates are estimated to be less than 4 mSv/h at contact with the BDL without the Insulator/Shield and less than 0.5mSv/h and 1/10th of the contact dose rate at 1metre without and with the insulator/shield, respectively. This would reduce the collective exposures for RIP Motor Maintenance to less than 0.7 person-mSv and 0.9 person-mSv for the BDL inspection. The work was summarised in (Ref. 142) ALARP Consideration on RPV Bottom Drain Line.

443. RO-ABWR-0034 was formally closed on 03rd March 2017. However, RO-ABWR-0035 relating to “Robust justification for the Materials Selected for the UK ABWR” remained open and required completion of the justification of the materials for construction of the BDL such that all risks are controlled SFAIRP. RO-ABWR-0035 was subsequently closed on19th September 2017.

4.5.1.1 Conclusion

444. I conclude from my Radiological Protection assessment of the BDL that Hitachi-GE has carried out a broad and robust analysis of the provision of the BDL and its benefits and detriments in relation to the overall risks and exposures to workers from normal and abnormal operations. I conclude that the analysis is commensurate with the expectations for GDA and the inclusion of the BDL with respect to radiological exposures is broadly acceptable.

4.5.2 Heating Ventilation and Air Conditioning

445. Further cross cutting issues included the design of the HVAC systems for the UK ABWR and specifically for the Reactor and Turbine Building. This assessment has been led by the Mechanical Engineering inspector and is primarily reported within the Mechanical Engineering report (Ref. 143).

446. Radiological Protection is concerned with respect to ventilation systems as they control radioactive material suspended in air by drawing it away from workers and either trapping the material on filters and or expelling it from the facility via a discharge point. Ventilation can be provided through general dilution or area ventilation and/or Local Extract Ventilation (LEV) where contaminants are removed away from workers local to the generation of the contamination.

447. It should be noted that the primary means of removing gaseous radioactive material from the UK ABWR operations and primary containment is via the off-gas system. This system removes H_2 and O_2 generated from radiolysis of reactor cooling water. This is further supported by the Stand-by Gas Treatment System which maintains a negative pressure in the secondary containment and filters gaseous effluents prior to discharges. It also plays a key role in processing gaseous effluent from the primary Containment Vessel, limiting radioactive discharges during normal and abnormal plant operation. Finally the HVAC controls temperature, pressure, humidity and airborne contamination to ensure the integrity of plant and equipment, provide acceptable working conditions for plant personnel, and limit offsite releases of airborne contaminants.

448. The HVAC system is made up of a number of sub-systems, 13 in total support the Reactor Building Emergency Diesel Generator Buildings (EDG/B), T/B, Heat Exchanger Building, Control Building specifically for Emergency Electrical Equipment Zone, MCR, RW/B, Service Building and Back Up Building. Each has supporting emergency roles.

449. It is important for Radiological Protection to ensure that the design principles applied to the HVAC are suitably developed and when applied support the overall contamination control for the plant. The design principles should minimise the radiological hazard and can be subject to EMIT such that exposures are restricted to SFAIRP. Further to this radioactive material is controlled appropriately and can be adequately accounted for. This aspect is very much related to radioactive waste and decommissioning areas specifically in relation to prevention of leakage and escape and accumulation of radioactive waste and accountancy.

450. HVAC has been subject to a number of relating ROs and RQs through the assessment process. Radiological Protection raised RO-ABWR-0064 which required Hitachi-GE to provide adequate arguments and evidence to support the claim that the design adequately considered contamination and radiation control through both fixed and moveable features. The Mechanical Engineering Inspector raised RO-ABWR-0017 (Ref. 144) to ensure the adequate demonstration of design process and application of the Nuclear Ventilation Codes and Standards. Further to this, RO-ABWR-0036 (Ref. 145) Demonstration that the approach taken to radioactive waste management reduces risks SFAIRP, and RO-ABWR-0075 (Ref.130) Robust demonstration that the design of the UK ABWR HVAC system has been adequately conceived and reduces risk SFAIRP, were raised.

451. ONR Cross cutting assessment of the HVAC system has led to a number of Assessment Findings which are primarily recorded in the Mechanical Engineering and Waste and Decommissioning Assessment Reports. Radiological Protection has provided input into this cross cutting assessment and its primary findings related to Filter Bank Design. This currently offers good practice in the form of safe change HEPA filters, which aids in reduction of potential for loss of containment and prevention of the spread of contamination within the working environment. The filter bank design, however, is primarily configured such that operators are exposed to a 270 degree radiation field during filter change. This is not seen as an ALARP configuration and does not meet my expectations.

452. Local Extract Ventilation (LEV) may be required during maintenance, fault scenarios and decommissioning. Currently the design relies on mobile extract units which may not always be appropriate. No safety justification has been provided on these systems within GDA. It is my expectation that future licensees will review the use of LEV post GDA.

453. In line with RO-ABWR-0064 there is little or no reference to local installed monitoring / sampling for radiation or airborne contamination in the HVAC Safety Case. This is

required to warn operators of adverse conditions within the filter or ventilation service rooms prior to entry. This is not considered best practice and does not meet ONR's expectations. This concern was included in assessment finding (AF-ABWR-RP-03) section 4.2.5.3.

454. Hitachi-GE has identified a number of options for improvement which could be applied to the design of the HVAC systems. These will be considered during the site specific phase.

4.5.2.1 Conclusion

455. I conclude from my assessment of the HVAC System as presented within the submitted documentation that it generally meets my expectations with respect to protection of the workforce from direct radiation and contamination.

456. There are two specific design features which do not meet my expectations as currently designed:

- The U-Shape design of the main HVAC filter bank will lead to what is potentially an increased and non-optimised exposure to the maintenance workers. This has been captured under AF-ABWR-RP-01
- A lack of installed airborne and radiation monitoring instrumentation linked to keep out warning lights and interlock systems. Currently the system relies on a locked door and administrative control of keys. This is lower down the hierarchy of controls and is not considered ALARP. This is to be addressed under AF-ABWR-RP-03.

4.5.3 Spent Fuel Pool Operational Working Environment and Environmental Factors

457. During Step 4 the EA received a response from Hitachi-GE to an RQ (RQ-ABWR-0247 (Ref. 146)) regarding the impact of the Spent Fuel Pool (SFP) temperature on discharges of Tritium from the SFP with respect to Best Available Technology (BAT). On review of this matter I noted the lack of attention to radiological protection with respect to operators within the SFP area and to the justification of exposure to tritium and other airborne species in relation to interactions of environmental factors such as SFP temperature, air temperature, and humidity and ventilation flow rates.

458. During my assessment of the response I noted Hitachi-GEs simplified assumptions in relation to ventilation flow rates/air changes, etc., would not provide a bounding range for the evaporation of tritium from the SFP and therefore the assertion that doses to workers and the public are ALARP. I agreed with EA to draft an appropriate query to discuss with Hitachi-GE.

459. During the development of ONR and EA's joint position I also discussed this issue with the Human Factors and Conventional Health and Safety Inspectors with respect to the operational working environment and the effect of comfort factors for the workforce in addition to the general environmental factors impacting evaporation rates and the BAT arguments.

460. Finally, following meetings with Hitachi-GE, EA and ONR Human Factors inspectors an RQ was agreed and issued requiring Hitachi-GE to provide additional information by which ONR and EA could assess the SFP storage requirements in relation to spent, nuclear fuel, control rods, stored materials, the requirements of workers' operational environment, exposures and discharges impacting the external environment and the public.

461. Hitachi-GE re-presented material including additional information. This included a more accurate assessment of the SFP temperature and other environmental factors. The

response also related the design to Hitachi-GE's Nuclear Safety and Environmental Design Principles. The SFP is estimated to be lower than the designed maximum value (65 ºC) during refuelling. The most significant challenge to environmental cooling within the spent fuel pool comes from a full core off-load when the pool can exceed normal heat load (but is controlled below 65 ºC). Hitachi-GE states that operational comfort is maintained at the operations floor. Hitachi-GE estimates the internal exposure from ^3H to a worker over 1 month during outage is 2.2E-02 mSv.

462. I and the Human Factors inspector considered the response to the RQ adequate to close out the query. I am content this component of exposure is optimised in relation to GDA.

4.5.4 Upper and Lower Drywell

463. During Step 4 the access and egress arrangements to the lower drywell (Fig 3). were investigated following discussions with the Fault Studies Inspector. The Fault Studies Inspector identified a period of perceived elevated risk to workers during both start-up and shut-down of the reactor due to the use of N_2 to inert the PCV as a means of preventing hydrogen combusting in an O_2 atmosphere.

464. Following the start-up of the UK ABWR with removal of the control rods, power is raised and so are temperature and pressure. Once a set level is reached valves controlling the steam are closed and the control rods reinserted to stop the nuclear reaction. Pressure is held and once the dose rates allow, workers enter the lower dry well and check for visible leaks from the RPV and associated structures and components e.g. FMCRD and RIP. Hitachi-GE reported that several teams of workers enter over a period of up to 120 minutes. Once the inspection is completed and with no visible signs of leaks the PCV is inerted using the atmosphere control system (AC) and the isolation valves are opened and control rods withdrawn. A similar but inverse process is carried out at Shutdown. At shutdown the control rods are inserted, the isolation valves are closed and the PCV is purged of N_2. A similar process is adopted with teams of workers entering over a period of up to 120 minutes to visibly inspect the structure and components for signs of leakage and identify any additional areas requiring repair or maintenance.

465. During these times the workers in the lower drywell are at potentially heightened risk of exposure to an anoxic atmosphere with dead areas of N_2 remaining, being exposed to elevated dose rates as the reactor has only recently been brought off power and will be in an environment of elevated temperature and humidity. This area is also considered a confined space.

466. Management of workers during this time will be an important aspect, ensuring their welfare, health and safety within this constrained environment along with the provision of rescue resources and equipment. This is specifically reflected within the Conventional Health and Safety Report (Ref. 147).

467. During Step 4 the Level 3 PSA and Radcons inspector identified through an OPEX review a number of events in light water reactors, primarily BWRs, where operators received overexposures. The inspector shared these events with other ONR inspectors to help inform the assessments. Included within this OPEX review was reference to US Nuclear Regulatory Commission guidance containing evidence of potential weaknesses in BWR design (Ref. 148). The primary concern of this guidance is with respect to potential drop or alignment of a spent fuel assemblie in proximity to the RPV wall and the Lower and more importantly Upper Drywell . Such a scenario could produce elevated and potentially fatal dose rates within the Upper Drywell.

468. The primary means of managing radiation and reducing dose rates to workers from sources within the RPV and reactor spent fuel and dryer/separator pits is through the

use of water as a shielding material. Secondary to this is the provision of a concrete shield. When an irradiated fuel assembly is adjacent to the RPV wall or RPV well bulkhead this water shield is negligible and therefore provides no shielding, leaving the only shielding as the RPV wall and potentially the Reactor Concrete Containment Vessel (RCCV) or bulkhead. There is a potential within the current UK ABWR design for significant dose rates within the upper dry well.

469. It is impractical to manage maintenance and outage arrangements to preclude access to all the dry well areas during fuel moves as this would lead to outages of such length and complexity to challenge the commercial viability of the operation.

470. An RQ was raised as a cross cutting RQ between: Radiation Protection, Fault Studies and PSA level 3. The RQ was devised to seek further evidence of Hitachi-GE's design in relation to demonstrating either the incredibility/credibility of the types of events which would lead to such dose rates and potential exposures or the arrangements to prevent or the mitigation of such events. The intention was to seek assurance the design met SAP FA2 (Ref. 6).

471. Hitachi-GE provided a response (Ref. 149). This response followed a four step process:

- Fault Identification.
- Evaluation of unmitigated consequences.
- Consideration of countermeasures for worker dose reduction.
- Conclusion.

472. The process identified four scenarios/cases listed below:

- Case 1: Fuel Assembly dropping and lying down in Reactor Well.
- Case 2: Fuel Assembly dropping to RPV Bottom.
- Case 3: Fuel Assembly staying above Outermost Edge of Reactor Core.
- Case 4: Fuel Assembly dropping to Top of Reactor Core.

473. The impact of these four scenarios/cases, were assessed against the modelled doses at four locations within the upper and lower drywell areas and the operating deck.

474. The document detailed the frequency of the fault as being of the order of $10^{-4}yr^{-1}$ or less and therefore is classed as an infrequent fault. Hitachi-GE argues that a potential drop or positioning of an irradiated fuel assembly onto or adjacent to the RPV bulkhead plate is bounding due to geometry and shielding provision. Hitachi further argues that due to the time of travel of the irradiated fuel assembly over the bulkhead plate, a distance of 2 metres, being 0.1 minutes in a total transit time of 3 minutes, the frequency is expected to be less than $10^{-5}yr^{-1}$.

475. The level of expected exposure was identified, the greatest being 500 mSv, equivalent to Target 4 BSL for initiating fault frequencies between $1x10^{-4}$ and $1x10^{-5}$ pa (Ref. 6).

476. Hitachi-GE then considered the necessary countermeasures for worker dose reduction in line with ERIC-PD. This process involved consideration of elimination of the scenario, reduction in the potential for a dropped assembly or reduction in dose rate if the event should happen, through to isolation, PPE and discipline. The considerations finally discussed in detail as potential options included:

- Temporary fuel transfer chute in the Reactor Well.
- Temporary shielding on the bulkhead plate.
- Rapid Evacuation.

477. Hitachi-GE concluded the following options were not reasonably practicable

- Increasing shielding by increasing the thickness of the Reactor Shielding Wall to protect workers in the Upper Drywell.
- Increasing the thickness of RPV for workers in both Upper and Lower Dry Wells.
- Increasing the thickness of the RCCV for workers in the Upper Dry Well.
- Installing additional permanent shielding for workers in both the Upper and Lower Dry Wells
- Increasing thickness of RPV pedestal for workers in the Lower Dry Well. .

478. I concur with Hitachi-GE's conclusion that the potential inclusion of a Temporary Fuel Transfer Chute and or use of Temporary Shielding on the bulkhead plate would seem reasonably practicable in reducing potential exposures. Hitachi-GE notes the need to assess the potential for other risks arising in relation to dropped loads over the reactor well area and the need to update the lifting schedule and assessment for dropped loads. This potential modification will still require development at the Site Licence stage and as such has been raised as an assessment finding.

AF-ABWR-RP-08: *The licensee shall assess the design of the UK ABWR fuel handling and transfer process to ensure adequate management of irradiated fuel from reactor core to its safe storage to ensure exposures to workers are reduced so far as is reasonably practicable, including those workers in the upper and lower drywell prior to the implementation of any evacuation requirements.*

479. The use of warning devices would also support defence in depth so as to inform workers in the area of a change in dose rates and instigate an evacuation from the area to aid in exposure mitigation. Evacuation should be seen as a defence in depth and as a means to reduce exposures so far as is reasonably practicable and beyond the BSO.

4.5.5 Evacuation Times

480. During the Radiological Protection Assessment a number of RQs were raised in relation to the identification of locations for workers and the identification of appropriate annunciation of emergency warnings/signals and the identified routes and times to evacuate. It became clear that Hitachi-GE was identifying overly optimistic evacuation times in relation to both unmitigated and mitigated fault scenarios such that the dose assessments were low.

481. Hitachi-GE was also identifying evacuation as a primary mitigation against the consequences of potential exposure scenarios rather than identifying means higher up the hierarchy of controls such as preventing the event occurring through the use of engineering solutions or applying engineering methods to mitigate the outcome, e.g. to isolate workers from the hazard.

482. ONR inspectors from Fault Studies, Human Factors and Radiological Protection raised a number of RQs and attended various level 4 meetings where appropriate influence was brought to bear on Hitachi-GE. Over the period of the step 4 assessment Hitachi-GE developed and strengthened its methodology and evidence in relation to evacuation times specifically with respect to those times being conservative so as to take account of workers' response to the event.

483. When developing a safety case applying an evacuation time of 1 to 2 minutes to an event and then assessing a potential exposure is likely to lead to an unrealistically low exposure and as such may lead to the early exclusion of an event from further consideration. To ensure an adequately realistic but bounding evacuation time Hitachi-GE has adopted a two-step approach. This has led to an initial assessment creating a bounding exposure and evacuation followed by a more realistic assessment taking into account mitigating factors and evacuation times. This provides a range of exposures.

In discussion and agreement with HF and Rad Cons inspectors have found this approach defendable and appropriate and in line with the expectations set out within NS-TAST-GD-045, Rev 3, Radiological Analysis – Fault Conditions (Ref.13).

4.5.6 Off-Gas Treatment system.

484. Section 18.3.1.2 of Chapter 18 of the UKABWR Generic PCSR (Ref. 150) states that "The OG is a key component of the ABWR design, which has the primary functions of maintaining the Main Condenser Vacuum, by extracting non-condensable gas, providing abatement of radioactive species prior to atmospheric discharge, and recombining radiolytic hydrogen and oxygen generated in the reactor.".

485. There are a number of aspects of the Off-Gas system which are of particular interest to Radiological Protection specifically with regard to minimisation of the source term, confinement and containment of radioactive material, provision of adequate shielding and protection of discharges to the workforce and public.

486. Chapter 18 presents a number of Safety Functional Claims which are related to the Off-Gas system with respect to Radiological Protection.

- OG SFC 4-7.1 "The OG minimises the dose to the worker during normal conditions."
- OG SFC 4-7.2 "The OG mitigates the dose to the worker in the event of the OG system failure."
- OG SFC 4-8.1 "The OG mitigates the release of gaseous radioactive substances to the environment in the event of the OG system failure."
- OG SFC 4-11.1 "The OG minimises the release of radioactivity to the environment during the start-up, power and shutdown operations."

487. The final claim is primarily of interest to NRW and the EA.

488. The Off-Gas system has been assessed in a cross cutting, multi-disciplinary manner with assessments being carried out with: NLR, Reactor Chemistry, Internal Hazards, Fault Studies and C&I. ONR's assessment of the Off-Gas system is delivered through the individual reports, with the radiological protection aspects being reported within this report.

489. During initial assessment of PCSR Rev B and its supporting documents I and inspectors from Reactor Chemistry, EA, Mechanical Engineering and NLR identified the need to draft and issue RO-ABWR-0073 (Ref.151). This RO required Hitachi-GE to provide a demonstration that the design of the UK ABWR off-gas system reduces risk SFAIRP.

490. My assessment is primarily based on the Hitachi-GE's response to RO-ABWR-0073 (Refs. 153, 155).

491. Hitachi-GEs safety case for the UK ABWR off-gas system is presented across a number of documents from PCSR Rev C, primarily Chapters 18 (Ref. 150), 20 (Ref. 31), 23 (Ref.152), through to the Basis of Safety Case (Ref. 154), Topic Reports on ALARP Assessment for the Off-Gas system (Ref. 153) and Technical Supporting Document on the OG ALARP Report (Ref.155) updated July 2017. The majority of Hitachi-GEs claims are presented in the Technical Support Document (Ref. 153).

492. Workers are not required to access the Off-Gas rooms during normal operation. The rooms are mainly categorised as R4 (0.5 mSv/h or above) (Ref. 154) on a radiation basis. Access Control to these rooms is via lock and key control, where access to the key is controlled under administrative arrangements. Rooms containing Off-Gas

equipment are provided with adequate shielding. Access is required to the Off-Gas equipment rooms during plant maintenance activities during shutdown. The primary source of radiological exposure within the Off-Gas system is ^{16}N and since the reactor is shutdown ^{16}N is no longer present in any quantity which would lead to radiological source of significance. The radioactive material remaining in the off-gas system would be fission products in the form of noble gases and other fission products entrained within the airborne component. The primary areas for worker exposures during maintenance come with respect to maintenance on the charcoal filter banks.

493. Hitachi-GE estimates the maintenance of workers from Off-Gas component maintenance during outages corresponds to 0.01% of the total average collective dose (this is calculated from dose information from its 5th, 6th, 7th and 9th periodical inspection for KK-7) (Ref. 155).

494. OPEX provided by Hitachi-GE indicates that maximum dose rates within Off-Gas Rooms during outage periods are less than 0.01mSvh^{-1}, the greatest dose rates being in the Turbine Gland Steam system (TGS) filter rooms (Ref. 155). The OG charcoal absorber room is less than 0.01mSvh^{-1}. The greatest exposure for workers involved in OG maintenance is related to the TGS filter with exposures estimated to be 0.08 person-mSv per outage.

495. With respect to contamination control, Hitachi-GE's Generic Design relies on the same exclusion principle to areas of elevated external dose rates during operation. Hitachi-GE recognises the potential for minor leakage during operation and state the design minimises this and that the contamination zoning take this into account.

496. Hitachi-GE presented evidence of an assessment of the Off-Gas system components and the potential for contamination spread during maintenance in the Radiological Protection Assessment (Ref. 155).

497. Hitachi-GE focuses on those components of the system housed within C3 areas, OPEX from KK-6/7 is limited to the Steam Jet Air Ejector (SJAE) and the Off-Gas recombiner. The TGS filter also lacks OPEX from KK-6/7 to back up Hitachi-GE claims, however, it is noted that the system is designed with Safe Change specifications and as such would reduce the potential for loss of containment. This in essence would meet ONR's expectations.

498. Further to this Hitachi-GE has designed the maintenance arrangements to include provision of temporary containment to aid in management of contamination. ONR's expectations are that contamination control is practiced at source and as such this provision is seen as a reasonably practicable approach. Further consideration of contamination control arrangements will be considered in more detail at Site Licensing.

499. Not withstanding the development of contamination control arrangements during the site specific phase, I sampled a number of operations and the locations proposed for installation of temporary containment, and temporary and permanent PPE/Clothing change locations. These included the areas identified for temporary control of those entering the drywell. I concluded from these reviews that adequate space was afforded within the UK ABWR generic design to provide for appropriate segregation of workers, materials and potential contaminated wastes.

4.5.6.1 Conclusion

500. In conclusion, exposures to workers during normal operations are precluded due to worker access to areas being restricted, primarily at the time of GDA through administrative key control and locked door measures, though not precluding the future use of radiation monitoring and provision of warning devices or interlocks as required under AF-ABWR-RP-03. Exposures during maintenance also seem reasonably low in

comparison to other maintenance activities on the reactor plant and information provided by Hitachi-GE (OPEX) gives confidence that this level of exposure should be realised.

501. Contamination control also seems to be considered appropriately at the GDA phase, specifically with the exclusion of the workforce during normal power operations and the appropriate design considerations including provision of space to introduce adequate temporary containment.

502. The Off-Gas System as detailed in GDA generally meets my expectations in relation to the operational radiological protection aspects for normal operational phases including those areas subject to inspection and maintenance.

4.6 Comparison with standards, guidance and relevant good practice

503. During my assessment I have compared Hitachi-GE's safety case, its claims, arguments and evidence provided against relevant standards, guidance and relevant good practice as detailed in section 2.1.3 and throughout the relevant sections of this report.

504. I have been cognisant of changes in standards and guidance as I have progressed through my assessment and I have communicated with Hitachi-GE to ensure they have been referring where appropriate to relevant changes in standards. RQ-ABWR-0498 (Ref. 156) was raised with the knowledge of potential changes in IRR99 following the adoption of Euratom Basic Safety Standards Directive 2013/59 (Ref. 157) and its implementation through revised legislation during 2018.

4.7 Overseas regulatory interface

505. ONR has formal information exchange agreements with a number of international nuclear safety regulators, and collaborates through the work of the International Atomic Energy Agency (IAEA) and the Organisation for Economic Co-operation and Development - Nuclear Energy Agency (OECD-NEA). This enables us to utilise overseas regulatory assessments of reactor technologies where they are relevant to the UK. It also enables the sharing of regulatory assessment findings which can expedite assessment and helps promote consistency.

506. ONR also represents the UK on the Multinational Design Evaluation Programme (MDEP). This seeks to:

- Enhance multilateral co-operation within existing regulatory frameworks.
- Encourage multinational convergence of codes, standards and safety goals.
- Implication of MDEP products in order to facilitate the licensing of new reactors, including those being developed by Gen IV International Forum.

507. In my radiological protection assessment, information was shared with the Swedish Nuclear Safety Authority (SSM) through electronic communication. I enquired on comparative examples of Swedish reactor design and aspects of my assessment in relation to BWRs.

508. Information was shared with MDEP through the delivery leads on specific aspects of the ABWR design.

509. The outputs from these interactions have given me the confidence that the challenges we are addressing on radiological protection in the UK are broadly similar to those in other countries. Whilst the way of dealing with the challenges is influenced by the regulatory regimes within countries, it is clear that all the regulators are working towards similar solutions for resolutions of these challenges.

4.8 Interface with Other Regulators

510. I have worked closely with the EA (also acting on behalf of NRW) through the whole of GDA. Future operators of the UK ABWR will require a permit from EA or NRW as advised by the EA assessment, to make discharges of radioactivity into the environment and off-site dispose of radioactive wastes. Working closely with the EA and NRW has been important since doses to members of the public during normal operations arise from discharges (regulated by NRW and EA as appropriate) and direct radiation originating within the site boundary (regulated by ONR). Also, within the workplace there are close interfaces between radiological protection and radioactive wastes regarding topics such as decontamination, decommissioning and waste handling.

511. Working closely with the EA meant raising joint ROs, holding joint meetings with Hitachi-GE, undertaking a number of visits and reviewing our respective assessments. I have ensured that ONR's TSC on radiological protection, TÜV SÜD Nuclear Technologies, was aware of the EA's roles and responsibilities when undertaking its work.

4.9 Assessment Findings

512. During my assessment 8 matters were identified for a future licensee to take forward in their site-specific safety submissions. Details of these are contained in Annex 5.

513. These matters do not undermine the generic safety submission and are primarily concerned with the provision of site specific safety case evidence, which will usually become available as the project progresses through the detailed design, construction and commissioning stages. These items are captured as assessment findings.

514. I have recorded residual matters as assessment findings if one or more of the following apply:

- Site specific information is required to resolve this matter.
- Resolving this matter depends on licensee design choices.
- The matter raised is related to operator specific features / aspects / choices.
- The resolution of this matter requires licensee choices on organisational matters.
- To resolve this matter the plant needs to be at some stage of construction / commissioning.

515. Assessment Findings are residual matters that must be addressed by the Licensee and the progress of this will be monitored by the regulator.

4.10 Minor Shortfalls

516. During my assessment 10 matters were identified as minor shortfalls in the safety case, but which are not considered serious enough to require specific action to be taken by the future licensee. Details of these are contained in Annex 6.

517. Residual matters are recorded as a minor shortfall if it does not:

- Undermine ONR's confidence in the safety of the generic design.
- Impair ONR's ability to understand the risks associated with the generic design.
- Require design modifications.
- Require further substantiation to be undertaken.

5 CONCLUSIONS

518. This report presents the findings of my Step 4 Radiological Protection assessment of the Hitachi-GE UK ABWR.

519. To conclude, I am broadly satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for Radiological Protection at this time, and these are referred to as appropriate in this report. As the GDA submission developed during Step 4, in response to my regulatory questions, amendments were made as appropriate to the PCSR Chapters (Refs. 24 through to 30) and its supporting references (which are listed in the Master Document Submission List (Ref. 158). I consider that from a Radiological Protection view point, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits beings secured . I consider that the current position with respect to the Radiological Protection Assessment of the UK ABWR design does not preclude the issue of a Design Acceptance Certificate.

520. Several assessment findings (Annex 5) were identified; these are for future licensees to consider and take forward in their site-specific safety submissions. These matters do not undermine the generic safety submission and they require licensee input/decision.

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Annex 1: Safety Assessment Principles

SAP No	SAP Title	Description
FP.3	Optimisation of protection	Protection must be optimised to provide the highest level of safety that is reasonably practicable.
FP.4	Safety assessment	Dutyholders must demonstrate effective understanding and control of the hazards posed by a site or facility through a comprehensive and systematic process of safety assessment.
FP.5	Limitation of risks to individuals	Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.
FP.6	Prevention of accidents	All reasonably practicable steps must be taken to prevent and mitigate nuclear or radiation accidents.
FP.7	Emergency preparedness and response	Arrangements must be made for emergency preparedness and response in case of nuclear or radiation incidents.
FP.8	Protection of present and future generations	People, present and future, must be adequately protected against radiation risks.
RP.1	Normal Operation (Planned Exposure Situations)	Adequate protection against exposure to radiation and radioactive substances should be provided in those parts of the facility to which access is permitted during normal operations
RP.2	Fault and Accident conditions (Emergency Exposure Situations)	Adequate protection against exposure to radiation and radioactive contamination should be provided in those parts of the facility that will need to be accessed during faults or as part of accident management. This should include prevention or mitigation of accident consequences.
RP.3	Designated areas	Where appropriate, designated areas should be further divided, with associated controls, to restrict and prevent the spread of radioactive material.
RP.4	Contaminated areas	Effective means for protecting persons entering and working in contaminated areas should be provided.
RP.5	Decontamination	Suitable and sufficient arrangements for decontaminating people, the facility, its [plant and equipment should be provided.
RP.6	Shielding	Where shielding has been identified as a means of restricting dose, it should be effective under all normal operation and fault conditions where it provides this safety function.

RP.7	Hierarchy of control measures	The dutyholder should establish a hierarchy of control measures to optimise protection in accordance with IRR99
NT.1	Assessment against targets	Safety cases should be assessed against the SAPs numerical targets for normal operational, design basis fault and radiological accident risks to people on and off site.
NT.2	Time at risk	There should be sufficient control of radiological hazards at all times.
NT.3	Applying the targets	When com paring the estimates submitted with the targets, inspectors should take account of the assumptions and limitations of the analysis used.
EKP.1	Inherent safety	The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
EKP.2	Fault Tolerance	The sensitivity of the facility to potential faults should be minimised
EKP.3	Defence in Depth	Nuclear facilities should be designated and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression.
EKP.4	Safety function	The safety function(s) to be delivered within the facility should be identified by a structured analysis.
EKP.5	Safety measures	Safety measures should be identified to deliver the required safety function(s).

Annex 2: Technical Assessment Guide

TAG Ref	TAG Title
NS-TAST-GD-002 Revision 4	Radiation Shielding
NS-TAST-GD-004 Revision 4	Fundamental Principles
NS-TAST-GD-005 Revision 8	Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)
NS-TAST-GD-038 Revision 6	Radiological Protection
NS-TAST-GD-043 Revision 2	Radiological Analysis Normal Operation
NS-TAST-GD-045 Revision 2	Radiological Analysis Fault Conditions

Annex 3: National and International Standards and Guidance

National and International Standards and Guidance
Safety of Nuclear Power Plants: Design. Safety Requirements. International Atomic Energy Agency (IAEA). Safety Standards Series No. NS-R-1. IAEA. Vienna. 2000. www.iaea.org.
Fundamental Safety Principles, Safety Fundamentals. International Atomic Energy Agency (IAEA) Safety Standards Series No. SF-1. IAEA, Vienna, 2006.
Safety of Nuclear Power Plants: Design. Specific Safety Requirements. International Atomic Energy Agency (IAEA). Safety Standards Series No. SSR-2/1. IAEA. Vienna. 2012.
Radiation Protection Aspects of Design for Nuclear Power Plants. IAEA Safety Standards Series, Safety Guide No. NS-G-1.13, International Atomic Energy Agency (IAEA) Vienna, 2005.
Safety of Nuclear Power Plants: Design Safety Standard - Specific Safety Requirements SSR-2/1 IAEA 2012.
Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards GSR Part 3 (Interim) IAEA 2011.
Western European Nuclear Regulators’ Association. Reactor Safety Levels for Existing Reactors September 2014.
WENRA Statement on Safety objectives for new nuclear power plants WENRA November 2010.
Safety of new NPP designs WENRA March 2013.
The Nuclear Installation Act.
The Energy Act.
The Ionising Radiations Regulations 1999.
The Ionising Radiations Regulations 1985.
The Radiation (emergency preparedness and public information) Regulations 2001.

Annex 4: Regulatory Issues / Observations

RI / RO Ref	RI / RO Title	Description	Date Closed	Report Section Reference
RO-ABWR-0014	UK ABWR Radiological Protection Safety Case: Project Plan and Delivery	The objective is to state ONR's expectations related to the development and delivery of the Radiological Protection safety case for the UK ABWR as part of the GDA submission. Hitachi-GE should develop and deliver a suitable and sufficient Radiological Protection safety case for UK ABWR in accordance with a detailed plan outlining specific Radiological Protection safety case tasks required to be completed and providing clarity on, and timings for, the deliverables. In response to the RO Hitachi-GE are requested to provide the UK ABWR Radiological Protection safety case documentation in a staggered but logical and timely fashion throughout GDA in accordance with the Radiological Protection safety case plan, and to keep the plan updated as the safety case and strategies evolve.	02/06/16	4.4.1
RO-ABWR-0064	Design approach to identification and provision of both permanent and temporary features necessary for the adequate control of radioactive contamination and over-exposure to radiation across the full lifetime of UKABWR	RO-ABWR-0064 is cross cutting and being raised to ensure the design of the UKABWR includes appropriate arrangements for both permanent and temporary features necessary for the adequate control of contamination and for the prevention of over-exposure to radiation are maintained through all phases and stages of operation of the UKABWR.	27/07/17	4.4.2
RO-ABWR-0065	Demonstration of adequate design and implementation of inherently safe techniques and structures to minimise radiation dose rates via through wall penetrations during all operating modes and for the lifetime of the facility, whilst being cognisant of design requirements relating to other discipline areas.	The objective of this RO is to state ONR's expectations related to the design for Shielding Penetrations and request Hitachi-GE to demonstrate how it will implement a design approach that meets ONR expectations for the design of the UKABWR.	02/05/17	4.4.3

Annex 5: Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RP-01	Context: ONR's GDA assessment identified there were omissions from Hitachi-GEs application and therefore demonstration of ALARP with respect to design of Solid and Liquid Waste Management Systems, the Heating Ventilation and Air Condition Systems (HVAC) and the Off-Gas Systems. Therefore the design of these systems cannot be said to be fully mature. This will leave a licensee with an incomplete design on which to base a safety case. AF: The licensee shall ensure the appropriate application of ALARP with respect to the GDA design of Solid and Liquid waste management, HVAC and Off-Gas systems. This shall include optimising these systems for decommissioning activity, minimising worker interventions for maintenance where reasonably practicable to do so and fully evaluating options identified in Topic Reports, such that the site specific design is optimised and risks, including radiological risks, to workers are reduced so far as is reasonably practicable.	4.2.2.2
AF-ABWR-RP-02	Context: ONR's GDA assessment identified there were opportunities for further optimisation of RPV opening and closing sequences to reduce overall risk to the workers so far as is reasonably practicable. Examples of areas for optimisation include: consideration of automation for Stud Tensioning/de-tensioning, nut removal, further development of remote decontamination and RPV water level control prior to removal of RPV head. It is necessary to develop the evidence supporting the argument that the proposed design of RPV head removal and reseating equipment and operational sequence is ALARP. Resolution of this aspect is important to reduce overall risk so far as is reasonably practicable.	4.2.2.3

	AF: To ensure adequate demonstration of ALARP the licensee shall determine the equipment and operational process required for RPV head removal and reseating that reduces radiological protection risks SFAIRP. This requires assessing options for automation, e.g. use of Multi Stud Tensioning (MST) devices. Along with this the licensee shall examine reasonably practicable options that enable the RPV to be filled with water and maintained at a higher level than currently achieved in J-ABWR, prior to removal of the RPV head to ensure dose rates are ALARP.	
AF-ABWR-RP-03	Context: The reference plant on which the UK ABWR design is based has limited provision for installed area radiological monitoring system (ARMS) linked to interlock systems or warning devices to inform or prevent inadvertent access to areas of potential high dose or contamination e.g. HVAC/off-gas filter rooms or liquid waste handling rooms. AF: The licensee is required to review the GDA requirement for the use of installed activity in air and direct radiation monitoring, such that appropriate locations are identified and the design facilitates adequate detection and signalling of data to relevant interlocks and warning devices to ensure exposure of workers is controlled so far as is reasonably practicable.	4.2.5.3
AF-ABWR-RP-04	Context: Hitachi-GE provided a number of documents in response to RO-ABWR-0064. One of these documents was entitled Contamination Control Philosophy. This report when reviewed as part of the suite of documents is considered adequate for GDA. Although the document provided a useful review of international OPEX and comparison against UK requirements the suite of documents did not provide a coherent overarching philosophy. This will be required to ensure the appropriate application of the hierarchy of controls to detailed design so as to demonstrate the control	4.4.2

	of contamination so far as is reasonably practicable. It is further noted that certain areas of contamination control are not as mature as ONR would expect, potentially due to the level of maturity of design and development of the overriding philosophy. AF: The Licensee shall develop an overarching philosophy regarding design for contamination control for the UK ABWR. This should ensure adequate source minimisation and control through the use of engineered features to minimise chronic and acute leakage. This should include but not be limited to use of HVAC and LEV, appropriate detection, and controls such as bunding and drainage.	
AF-ABWR-RP-05	Context: ONR identified a number of processes/procedures, within Hitachi-GE design documents, where activities were duplicated and opportunities were potentially available for dose and waste reduction. These include activities such as decontamination. The Site specific phase provides opportunity for the licensee to review the design and associated processes and procedures to ensure risks including radiological risks are reduced so far as is reasonably practicable. AF: The Licensee shall review the design for opportunities to minimise duplicate tasks within work programmes required for decontamination tasks taking due account of potential detriments (worker dose, secondary waste creation) to ensure risks are controlled so far as is reasonably practicable.	4.4.2
AF-ABWR-RP-06	Context: The reference plant on which the UKABWR design is based makes limited provision for the inclusion of installed LEV to support intrusive tasks. This has the potential to challenge the control of contamination within the generic design.	4.4.2

	AF: The Licensee shall develop an assessment on the use of LEV based around the specific hazards posed and develop the necessary controls and arrangements. This is required as the implementation of fixed engineering controls should take precedence over mobile units due to the inherent nature of management and use of such units.	
AF-ABWR-RP-07	Context: ONR's assessment identified the use of Lead wool for closing the annular gaps through penetrations within the UK ABWR design. During the process ONR also identified Hitachi-GEs design philosophy did not include joggling/inherently safe designs for penetrations. During the process Hitachi-GE reviewed the design and provided evidence of penetration design philosophy, design rule and examples of the application of this philosophy and rule. This included the potential use of inherently safe offset penetration (joggle) design. AF: The licensee shall develop penetration design in line with the proposed philosophy and design rule, applying inherently safe designs for penetrations ensuring use of lead wool is minimised or where reasonably practicable removed to ensure risks to employees are reduced so far as is reasonably practicable.	4.4.3
AF-ABWR-RP-08	Context: ONR's GDA assessment identified omissions within the assessment of potential radiological exposures during a failure in movement of irradiated fuel from the core to the Spent Fuel Pool. Hitachi-GE provided a revised assessment to close this gap and identified a number of options which require consideration through appropriate optioneering.	4.5.4

	AF: The licensee shall assess the design of the UK ABWR fuel handling and transfer process to ensure adequate management of irradiated fuel from reactor core to its safe storage to ensure exposures to workers are reduced so far as is reasonably practicable, including those workers in the upper and lower drywell prior to the implementation of any evacuation requirements.	

Annex 6: Minor Shortfalls

Minor Shortfall Number	Minor Shortfall Finding	Report Section Reference
MS-UKABWR-RP-01	Opportunity exists to better align the designation of areas for the UK ABWR with Relevant Good practice during site licensing. The Licensee should review the current radiological zoning criteria (specifically for supervised areas) to be consistent with current UK RGP for operational NPP's.	4.2.3.2
MS-UKABWR-RP-02	The licensee should review the approach applied to supervised areas with respect to contamination control so as to ensure any contamination is managed SFAIRP.	4.2.3.2
MS-UKABWR-RP-03	The Licensee should consider reviewing zoning areas which are based solely on relevant OPEX information and incorporate modelling to ascertain if appropriate zoning has been achieved.	4.2.3.3
MS-UKABWR-RP-04	The Licensee should review the zoning arrangements in the Turbine Building and identify whether additional engineering controls are required to restrict access between zones.	4.2.3.4
MS-UKABWR-RP-05	As Hitachi-GE has not provided all the GDA radiological and contamination zoning for the nuclear island (specifically the service building and parts of the Radwaste building), the licensee shall provide appropriate radiological and contamination zoning for all relevant buildings for the UK ABWR.	4.2.3.4
MS-UKABWR-RP-06	As it is common practice to have a colour- coded zoning maps, the Licensee should consider implementing a colour coded zoning maps to help prevent radiation and contamination classification 'islands' from forming.	4.2.3.4
MS-UKABWR-RP-07	The Licensee should consider presenting detailed shielding data in one reference document, in order to facilitate shielding calculations during the site specific phase, to help prevent omissions within the licensee's documentation.	4.2.4.6
MS-UKABWR-RP-08	The Licensee should further develop the ALARP evaluation for the CST and SPT to better demonstrate that there is adequate shielding to reduce public dose SFAIRP.	4.2.6.2

MS-UKABWR-RP-09	The licensee should provide ONR with an appropriate estimate of the maximum individual and collective dose for the UK ABWR, taking into account updated conversion factors and other dose reduction tools, including appropriate consideration of conservatism as necessary in line with RGP. This should also include the facilities that were in a concept design at GDA stage (SFIS, LLW and ILW facilities).	4.2.7.2
MS-UKABWR-RP-10	The Licensee when completing dose assessment for all aspects of the UK ABWR should ensure appropriate account is taken for all components of exposure. Where a component is not believed to be significant, justification should be provided.	4.2.7.2