Office for Nuclear Regulation

New Reactors Division – Generic Design Assessment

Step 4 Assessment of Radiological Protection and Criticality for the UK HPR1000 Reactor

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EXECUTIVE SUMMARY

This report presents the findings of my assessment of the Radiological Protection and Criticality aspects of the UK HPR1000 reactor design undertaken as part of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA). My assessment was carried out using the Pre-Construction Safety Report (PCSR) and supporting documentation submitted by the Requesting Party (RP).

The objective of my assessment was to make a judgement, from a Radiological Protection and Criticality perspective, on whether the generic UK HPR1000 design could be built and operated in Great Britain, in a way that is acceptably safe and secure (subject to site-specific assessment and licensing), as an input into ONR's overall decision on whether to grant a Design Acceptance Confirmation (DAC).

The scope of my GDA assessment was to review the safety aspects of the generic UK HPR1000 design by examining the claims, arguments and supporting evidence in the safety case. My Step 4 of GDA assessment built upon the work undertaken in GDA Steps 2 and 3 which enabled a judgement to be made on the adequacy of the Radiological Protection and Criticality information contained within the PCSR and supporting documentation.

My assessment focussed on the following aspects of the generic UK HPR1000 safety case:

- Normal operation – source terms
- Radiation shielding
- Worker dose assessment
- Public dose assessment
- Radiation and contamination zoning
- Radiation and contamination monitoring
- Post-accident accessibility
- Criticality safety for fuel storage
- Demonstration that relevant risks have been reduced As Low As Reasonably Practicable (ALARP)
- Consolidated safety case
- Comparison with defined standards, guidance and Relevant Good Practice (RGP)

The conclusions from my assessment are:

- The RP has provided appropriate arguments and evidence to corroborate the claims made within the PCSR Chapter 22 Radiological Protection.
- The RP has provided appropriate arguments and evidence to corroborate the claims made within the PCSR Chapter 5 Reactor Core from a Criticality safety for fuel storage perspective.
- The RP has provided appropriate arguments and evidence to corroborate the claims made within the PCSR Chapter 32 Emergency Arrangements.
- From the assessment the RP meets the expectations of relevant defined standards, guidance, Safety Assessment Principles (SAPs) and Technical Assessment Guides (TAGs) and RGP identified within the report.

These conclusions are based upon the following factors:

- A detailed and in-depth technical assessment, on a sampling basis, of the full scope of safety submissions at all levels of the hierarchy of the generic UK HPR1000 safety case documentation.

- Independent information, reviews, and analysis of key aspects of the generic safety case undertaken by Technical Support Contractors (TSCs).
- Detailed technical interactions with the RP, alongside the assessment of the responses to the substantial number of Regulatory Queries (RQs) and three Regulatory Observations (ROs) raised during the GDA.

Several matters also remain, which I judge are appropriate for a licensee to consider and take forward in its site-specific safety submissions. These matters do not undermine the generic UK HPR1000 design and safety submissions but are primarily concerned with the provision of site-specific safety case evidence which will become available as the project progresses through the detailed design, construction, and commissioning stages. These matters have been captured in nine Assessment Findings.

Overall, based on my assessment undertaken in accordance with ONR's procedures, 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 HPR1000 design. I recommend that from a Radiological Protection and Criticality perspective a DAC may be granted.

LIST OF ABBREVIATIONS

AF	Assessment Finding
ALARP	As Low As Reasonably Practicable
ANSI	American Nuclear Standard Institute
BAT	Best Available Technique
BFX	Fuel Building
BMS	Business Management System
BNX	Nuclear Auxiliary Building
BQF	Spent Fuel Interim Storage Facility
BQZ	Interim Storage Facility for Intermediate Level Waste
BRX	Reactor Building
BSA	Safeguard Building A
BSB	Safeguard Building B
BSC	Safeguard Building C
BSL	Basic Safety Level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
BWX	Radioactive Waste Treatment Building
CSC	Criticality Safety Criteria
CGN	China General Nuclear Power Corporation Ltd
DAC	Design Acceptance Confirmation
DBA	Design Basis Accidents
DCFs	Dose Conversion Factors
DEC-A	Design Extension Condition A
DR	Design Reference
EBA [CSBVS]	Containment Sweeping and Blowdown Ventilation System
EMIT	Examination. Maintenance, Inspection and Testing
EOP	Emergency Operating Procedure
EPD	Electronic Personal Dosimeter
EPR16	The Environmental Permitting Regulations (England and Wales) 2016
ERICPPE	Eliminate, Reduce, Isolate, Control and Personal Protective Equipment
FA	Fuel Assembly
GDA	Generic Design Assessment
GNI	General Nuclear International Ltd
GNSL	General Nuclear System Ltd.
HSG253	Health and Safety Guidance No. 253
HVAC	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
ICIA	In-Core Instrumentation Assembly

ICRP	International Commission on Radiological Protection
ILW	Intermediate Level Waste
IRR17	Ionising Radiations Regulations 2017
KRT [PRM]	Plant Radiation Monitoring
LC	License Condition
LOCA	Loss Of Cooling Accident
MCR	Main Control Room
NFA	New Fuel Assembly
NLR	Nuclear Liabilities Regulations
NPP	Nuclear Power Plants
NT	Numerical Targets
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PAA	Post-Accident Accessibility
PCER	Pre-construction Environmental Report
PCSR	Pre-construction Safety Report
PHE-CRCE	Public Health England Centre for Radiation, Chemicals and Environmental Hazards
PPE	Personal Protective Equipment
PWR	Pressurised Water Reactor
QA	Quality Assurance
ROA	Regulatory Observation Action
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pumps
RCV [CVCS]	Chemical and Volume Control System
REPPIR19	Radiation (Emergency Preparedness and Public Information) Regulations 2019
RGP	Relevant Good Practice
RPE	Respiratory Personal Equipment
RPE [VDS]	Vent and Drain System
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SA	Severe Accident
SAMG	Severe Accident Management Guidelines
SAP	Safety Assessment Principle(s)
SDR	Shielding Design Report
SFA	Spent Fuel Assembly

SFAIRP	So Far As Is Reasonably Practicable
SFCCM	Spent Filter Cartridge Change Machine
SG	Steam Generator
SoDA	(Environment Agency's) Statement of Design Acceptability
SQEP	Suitably Qualified and Experienced Personnel
SSC	Structures, Systems and Components
S/U	Sensitivity/Uncertainty
TAG	Technical Assessment Guide(s)
TSC	Technical Support Contractor
UK WPC	United Kingdom Working Party on Criticality
V&V	Verification and Validation
WENRA	Western European Nuclear Regulators' Association

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1 INTRODUCTION

1.1 Background

1. This report presents my assessment conducted as part of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA) for the generic UK HPR1000 design within the topic of Radiological Protection and Criticality.

2. The UK HPR1000 is a pressurised water reactor (PWR) design proposed for deployment in the UK. General Nuclear System Ltd (GNSL) is a UK-registered company that was established to implement the GDA on the generic UK HPR1000 design on behalf of three joint Requesting Parties (RP), i.e. China General Nuclear Power Corporation (CGN), EDF SA and General Nuclear International Ltd (GNI).

3. GDA is a process undertaken jointly by the ONR and the Environment Agency. Information on the GDA process is provided in a series of documents published on the joint regulators' website (www.onr.org.uk/new-reactors/index.htm). The outcome from the GDA process sought by the RP is a Design Acceptance Confirmation (DAC) from ONR and a Statement of Design Acceptability (SoDA) from the Environment Agency.

4. The GDA for the generic UK HPR1000 design followed a step-wise approach in a claims-argument-evidence hierarchy which commenced in 2017. Major technical interactions started in Step 2 which focussed on an examination of the main claims made by the RP for the UK HPR1000. In Step 3, the arguments which underpin those claims were examined. The Step 2 reports for individual technical areas, and the summary reports for Steps 2 and 3 are published on the joint regulators' website. The objective of Step 4 was to complete an in-depth assessment of the evidence presented by the RP to support and form the basis of the safety and security cases.

5. The full range of items that form part of my assessment is provided in ONR's GDA Guidance to Requesting Parties (Ref. 1). These include:

   - Consideration of issues identified during the earlier Step 2 and 3 assessments;
   - Judging the design against the Safety Assessment Principles (SAPs) (Ref. 2) and whether the proposed design ensures risks are As Low As Reasonably Practicable (ALARP);
   - Reviewing details of the RP's 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 a more detailed engineering design;
   - Assessing arrangements for ensuring and assuring that safety claims and assumptions will be realised in the final as-built design, and
   - Resolution of identified nuclear safety and security issues or identifying paths for resolution.

6. The purpose of this report is therefore to summarise my assessment in the Radiological Protection and Criticality topic which provides an input to the ONR decision on whether to grant a DAC, or otherwise. This assessment was focused on the submissions made by the RP throughout GDA, including those provided in response to the Regulatory Queries (RQs) and Regulatory Observations (ROs) I raised. The ROs issued to the RP are published on the GDA's joint regulators' website, together with the corresponding resolution plans.

1.2 Scope of this Report

7. This report presents the findings of my assessment of the Radiological Protection and Criticality safety for fuel storage aspects of the generic UK HPR1000 design undertaken as part of GDA. I carried out my assessment using the Pre-construction Safety Report (PCSR) (Ref. 3) and supporting documentation submitted by the RP. My assessment was focussed on considering whether the generic safety case provides an adequate justification for the generic UK HPR1000 design, in line with the objectives for GDA.

1.3 Methodology

8. The methodology for my assessment follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (Ref. 4).

9. My assessment was undertaken in accordance with the requirements of ONR's How2 Business Management System (BMS). ONR's SAPs (Ref. 2), together with supporting Technical Assessment Guides (TAG) (Ref. 4), were used as the basis for my assessment. Further details are provided in Section 2. The outputs from my assessment are consistent with ONR's GDA Guidance to RPs (Ref. 1).

2 ASSESSMENT STRATEGY

10. The strategy for my assessment of the Radiological Protection and Criticality safety for fuel storage aspects of the generic UK HPR1000 design and safety case is set out in this section. This identifies the scope of the assessment and the standards and criteria that have been applied.

2.1 Assessment Scope

11. A detailed description of my approach to this assessment can be found in assessment plan ONR-GDA-UKHPR1000-AP-19-016 Rev 1 (Ref. 5).

12. I considered all of the main submissions within the remit of my assessment scope, to various degrees of breadth and depth. I chose to concentrate my assessment on those aspects that I judged to have the greatest safety significance, or where the hazards appeared least well controlled. My assessment was also influenced by the claims made by the RP, my previous experience of similar systems for reactors and other nuclear facilities, and any identified gaps in the original submissions made by the RP. A particular focus of my assessment has been the RQs and ROs I raised because of my on-going assessment, and the resolution thereof.

13. Radiological protection impact on source term during decommissioning has been considered as part of minimisation of source term during this assessment. Other aspects relating to radiological protection will be considered in the Decommissioning assessment report (ONR-NR-AR-21-015) (Ref. 6) and are not covered within my report.

2.2 Sampling Strategy

14. In line with ONR's guidance (Ref. 4), I chose a sample of the RP's submissions to undertake my assessment. In doing so, I focussed on matters which I judged to be the most safety significant, where significant design or safety case changes may have been needed, or where there was a potential for a significant matter to be revealed that could have prevented ONR issuing a DAC. A particular focus of my assessment has therefore been the RQs and ROs that I raised as a result of my assessment, and the subsequent resolution of these matters.

15. Consistent with my Step 4 of GDA Radiological Protection and Criticality assessment plan (Ref. 5), my assessment focussed on whether the generic UK HPR1000 safety case provides an adequate justification for the generic UK HPR1000 design, for those aspects relating to Radiological Protection and Criticality. The main themes I considered throughout Step 4 of GDA were:

- Designation and zoning of radiation and contamination areas;
- review of the ALARP process undertaken by the RP (holistic and specific assessment aspects);
- use of Operational Experience (OPEX) for source terms;
- occupational radiation exposure;
- public radiation exposure from direct shine;
- criticality safety of out-of-core fuel handling and storage;
- radiation shielding design, and
- review of the Radiological Protection and Criticality aspects of the safety case to be compliant with current UK standards and guidance.

16. I also followed up on a key item identified in my Step 3 of GDA Radiological Protection assessment as detailed in my Step 4 of GDA Radiological Protection and Criticality assessment plan (Ref. 5). This was that dose to the members of the public due to

direct radiation shine are based solely on calculation, which is described in detail in Section 4 of this report.

2.3 Out of Scope Items

17. The following items were outside the scope of my Step 4 of GDA Radiological Protection and Criticality assessment:

- Review of accident source terms.
- A detailed ALARP assessment of the generic UK HPR1000 waste building design for direct radiation shine.
- Assessment of Environmental monitoring.
- Review of the habitability of buildings other than the Main Control Room (MCR) and technical support room used within an emergency.
- My assessment of the adequacy of the criticality safety of fuel storage was confined to assessment of the designs of the New Fuel Assembly (NFA) store and Spent Fuel Assembly (SFA) pool, in line with my Step 4 of GDA Radiological Protection and Criticality assessment plan (Ref. 5). Loading and unloading of Fuel Assemblies (FA) to the NFA store, SFA pool and Spent Fuel Interim Storage Facility (BQF), and other storage-related operations for nuclear fuel (such as post-accident recovery, transfer to and from the reactor core, storage of damaged or non-standard fuels) have not been fully defined by the RP during Step 4 of GDA and are therefore outside the scope of my criticality safety for fuel storage safety assessment.

2.4 Standards and Criteria

18. The relevant standards and criteria adopted within this assessment are principally the SAPs (Ref. 2), TAGs (Ref. 4), relevant national and international standards, and relevant good practice informed from existing practices adopted on nuclear licensed sites in Great Britain. The key SAPs and any relevant TAGs, national and international standards and guidance are detailed within this section. Relevant good practice (RGP), where applicable, is cited within the body of the assessment.

2.4.1 Safety Assessment Principles

19. The SAPs (Ref. 2) constitute the regulatory principles against which ONR judge the adequacy of safety cases. The SAPs applicable to Radiological Protection and Criticality are included within Annex 1 of this report.

20. The key SAPs applied within my assessment are Fundamental Principle, FP3 and Radiological Protection RP1, RP2, RP3, RP4, RP5, RP6 and RP7.

2.4.2 Technical Assessment Guides

21. The following TAGs were used as part of this assessment (Ref. 4):

- NS-TAST-GD-005, (Rev 11), ‘ONR Guidance on the Demonstration of ALARP’
- NS-TAST-GD-038, (Rev 9), ‘Radiological Protection’
- NS-TAST-GD-043, (Rev 6), ‘Radiological Analysis Normal Operation’
- NS-TAST-GD-002, (Rev 8), ‘Radiation Shielding’
- NS-TAST-GD-041, (Rev 7), ‘Criticality Safety’
- NS-TAST-GD-096, (Rev 0), ‘Guidance on Mechanics of Assessment’
- NS-TAST-GD-051, (Rev 7), ‘The Purpose, Scope and Content of Nuclear Safety Cases’.

2.4.3 National and International Standards and Guidance

22. The following standards and guidance were used as part of this assessment:

- International Atomic Energy Agency (IAEA) guidance – 'Radiation Protection Aspects of Design for Nuclear Power Plants' (Ref. 7)
- 'Occupational Radiological Protection Principles and Criteria for designing New Nuclear Power Plants'. Nuclear Energy Agency, (Ref. 8)
- Western European Nuclear Regulators Association (WENRA), 'Reactor Safety Reference Levels for Existing Reactors' (Ref. 9)
- 'Ionising Radiations Regulations 2017' (IRR17) (Ref. 10)
- 'Radiation (Emergency Preparedness and Public Information) Regulations 2019' (REPPIR19) (Ref. 11).

23. There are both IAEA standards and WENRA Reference Levels of relevance and it should be noted that the latest version of the SAPs (Ref. 2) has been benchmarked against both IAEA and WENRA guidance at the time of publication.

2.5 Use of Technical Support Contractors

24. It is usual in GDA for ONR to use technical support contractors (TSC) to provide access to independent advice and experience, analysis techniques and models, and to enable ONR inspectors to focus on regulatory decision making.

25. Table 1 below sets out the areas in which I used TSCs to support my assessment. I required this support to provide additional capacity and access to independent technical support and access to analysis techniques.

Table 1: Work Packages Undertaken by the TSC

Number	Description
1	An independent review/assessment of a sample of the RP's safety case submissions which identify and justify the radiation shielding provisions for the UK HPR1000 design. This involved: ■ Undertaking independent radiation shielding modelling; ■ advising ONR on the adequacy of the radiation shielding design proposed for UK HPR1000; and ■ advising ONR on the adequacy of the RP's radiological protection arrangements, with regard to restricting radiation exposures of workers and the public, so far as is reasonably practicable (SFAIRP*).
2	An independent review/assessment of the criticality safety for fuel storage safety case submissions, which demonstrate criticality safety for fuel storage safety for the UK HPR1000 SFP and NFA design. Advising ONR on the adequacy of criticality safety for fuel storage safety. This involved undertaking independent criticality modelling.

26. The TSC output is referred to where required within my assessment. Whilst the TSC undertook detailed technical reviews, this was done under my direction and close supervision. The regulatory judgment on the adequacy, or otherwise, of the generic UK HPR1000 safety case in this report has been made exclusively by ONR.

* *The term SFAIRP is used interchangeably with ALARP throughout this report and essentially meaning the same definition.*

2.6 Integration with Other Assessment Topics

27. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot be carried out in isolation as there are often issues that span multiple disciplines. I have therefore worked closely with a number of other ONR inspectors and the Environment Agency to inform my assessment. The key interactions were:

- Environment Agency, on areas related to the minimisation of source term and dose to members of the public due to direct radiation shine.
- Reactor Chemistry, on areas related to the characterisation and minimisation of source term.
- Nuclear Liabilities Regulation (NLR), on areas related to the minimisation of radioactive waste, radiological protection aspects of decommissioning and requirements for spent fuel storage on site.
- Mechanical Engineering, on areas related to Health and Safety Guidance No. 253 (HSG253) compliance and ventilation.
- Fault Studies and Severe Accidents, on areas related to source term for Design Basis Accidents (DBA) and ONR SAP Numerical Targets (NT) 4, 5 and 6 in relation to Post Accident Accessibility (PAA).
- Fuel and Core, on areas related to the criticality safety for fuel storage assessment for the SFP.

2.7 Overseas Regulatory Interface

28. ONR has formal information exchange agreements with a number of international nuclear safety regulators and collaborates through the work of the IAEA and the Organisation for Economic Co-operation and Development Nuclear Energy Agency. 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 assessments, which can expedite assessment and helps promote consistency.

3 REQUESTING PARTY'S SAFETY CASE

3.1 Introduction to the Generic UK HPR1000 Design

29. The generic UK HPR1000 design is described in detail in the PCSR. It is a three-loop PWR designed by CGN using the Chinese Hualong technology. The generic UK HPR1000 design has evolved from reactors which have been constructed and operated in China since the late 1980s, including the M310 design used at Daya Bay and Ling'ao (Units 1 and 2), the CPR1000, the CPR1000+ and the more recent ACPR1000. The first two units of CGN's HPR1000, Fangchenggang Nuclear Power Plant Units (NPP) Units 3 and 4, are under construction in China and Unit 3 is the reference plant for the generic UK HPR1000 design. The design is claimed to have a lifetime of at least 60 years and has a nominal electric output of 1,180 MW.

30. The reactor core contains zirconium clad uranium dioxide (UO_2) fuel assemblies and reactivity is controlled by a combination of control rods, soluble boron in the coolant and burnable poisons within the fuel. The core is contained within a steel Reactor Pressure Vessel (RPV) which is connected to the key primary circuit components, including the Reactor Coolant Pumps (RCPs), Steam Generators (SGs), pressuriser and associated piping, in the three-loop configuration. The design also includes a number of auxiliary systems that allow normal operation of the plant, as well as active and passive safety systems to provide protection in the case of faults, all contained within a number of dedicated buildings.

31. The Reactor Building (BRX) houses the reactor and primary circuit and is based on a double-walled containment with a large free volume. Three separate Safeguard Buildings (BSA, BSB and BSC) surround the BRX and house key safety systems and the MCR. The Fuel Building (BFX) is also adjacent to the reactor and contains the fuel handling and short-term storage facilities. Finally, the Nuclear Auxiliary Building (BNX) contains a number of systems that support operation of the reactor. In combination with the diesel, personnel access and equipment access buildings, these constitute the nuclear island for the generic UK HPR1000 design.

3.2 The Generic UK HPR1000 Safety Case

32. In this section I provide an overview of the radiological protection and criticality safety for fuel storage aspects as well as emergency arrangements of the generic UK HPR1000 safety case as provided by the RP during GDA. The primary PCSR submission that I assessed is PCSR Chapter 22, which covers radiological protection (Ref. 3). The details of the technical content of the documentation and my assessment of its adequacy are reported in the subsequent sections of my report.

3.2.1 Safety Case Structure

33. The generic UK HPR1000 safety case follows a defined claims, arguments and evidence structure. In the route map for PCSR Chapter 22 (Ref. 3) the fundamental objective of the UK HPR1000 is stated to be:

> "Fundamental Objective: The Generic UK HPR1000 could be constructed, operated, and decommissioned in the UK on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the environment."

34. To underpin the fundamental objective, high-level safety claims are set out in 'PCSR Chapter 1 Introduction' (Ref. 12). The radiological protection and criticality claims derived from high-level claim 3[†].

[†] *The claims relating to radiological protection made by the RP were assessed during step 2 and step 3 and I found these to be acceptable.*

"Claim 3: The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable."

35. The supporting chapter level claims and arguments specific to radiological protection (including criticality safety for fuel storage) are set out in 'PCSR Chapter 22 Radiological Protection' (Ref. 3) The key radiological protection claims, and sub-claims are as follows:

- "Claim 3.4: The safety assessment shows that the nuclear safety risks are ALARP."
- "Claim 3.4.4: The risk to workers and members of the public from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP."
    - "Sub-claim 3.4.4.SC22.1: The risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP."
    - "Sub-claim 3.4.4SC22.2: The risk to members of the public from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP."
- "Claim 3.4.5: The risk to workers and members of the public from the potential harmful effects of ionising radiation resulting from fault and accident conditions complies with UK legal requirements and is ALARP."
    - "Sub-claim 3.4.5.SC22.1: The risk to workers mitigating fault/accident conditions complies with UK legal requirements and is ALARP."

36. The RP presented evidence in support of the radiological protection claims, arguments and sub-arguments in a "route map" in PCSR Chapter 22 (Ref. 3). This provided information on the submissions prepared to support the claims and sub claims listed above. I have referred to the relevant claims and arguments in my report, within Section 4 in forming a judgement on their adequacy and substantiation.

3.3 Safety Case Documentation

37. A description of the RP's chosen documentation hierarchy for the generic UK HPR1000 safety case in regard to radiological protection, is provided in 'Production Strategy for Radiological Protection' (Ref. 13). The overall documentation structure is divided into four tiers.

38. Tier 1 comprises Chapter 22 of the PCSR (Ref. 3) which the RP considers to be the top-level overview document for the radiological protection safety case. PCSR Chapter 22 presents and develops the radiological protection related claims and arguments and is supported by detailed documentation presenting the underlying evidence.

39. Tier 2 documentation supports and substantiates the claims in PCSR Chapter 22 and provides the link between it and further detailed evidence presented in tier 3 and tier 4 documentation.

40. Tier 3 submissions include detailed evaluation and analysis documents which provide further support and substantiation to the radiological protection claims and arguments set out in PCSR Chapter 22 and tier 2 documentation.

41. Tier 4 submissions include RO and RQ responses, presentations, production strategy and technical meeting minutes.

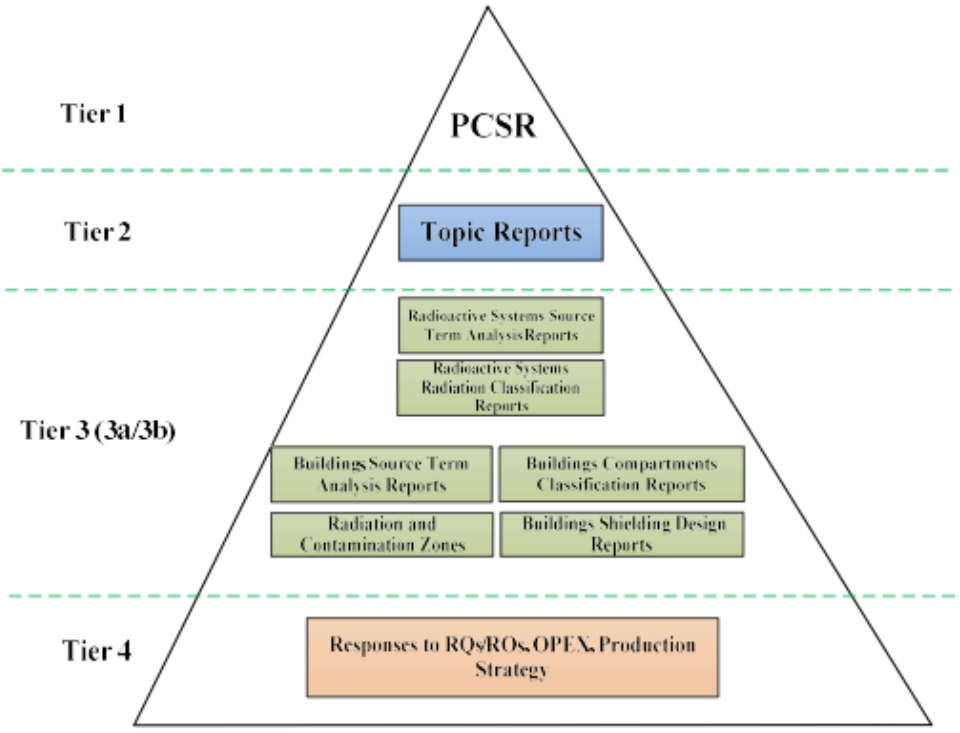

Figure 1: Documentation Hierarchy of Radiological Protection (Ref. 13)

42. The documentation forming the radiological protection safety case is extensive and is set out in the 'Master Document Submission List' (Ref. 14). The key submissions are referenced and described in Section 4 of my report.

3.4 Radiological Protection PCSR Chapter 22

43. The PCSR is a top-level document in the UK HPR1000 generic safety case which, together with the Pre-Construction Environmental Report (PCER) and the Generic Security Report, aims to demonstrate that the design meets UK safety, security and environmental requirements and that relevant risks are reduced SFAIRP. The PCSR consists of 33 chapters covering a general description of the plant, generic site characteristics and design principles, and chapters dedicated to key systems and technical topics.

44. Chapter 22 of the PCSR (Ref. 3) covers the radiological protection aspects of the generic UK HPR1000 safety case for normal operation, which is start-up, power operation, shutdown, maintenance, testing and refuelling. It also describes the measures in place to reduce exposure of workers to radiation and contamination during accident mitigation. The chapter identifies the radiological hazards associated with normal operation of the generic UK HPR1000 design and defines the radioactive sources during normal operation that have been considered.

45. PCSR Chapter 22 (Ref. 3) defines the RP strategy to ensure that the exposure to radiation is ALARP, and the protection measures that have been considered in the design to protect against direct radiation and radioactive contamination. Together with its supporting references, the RP demonstrated that the radiation doses to workers during normal operation comply with the UK legal requirements and are ALARP. This chapter is divided into several topic areas.

46. The RP provided a high-level summary of the applicable codes and standards which relate to radiological protection, for example IRR17 (Ref. 10). The RP used these

applicable codes and standards to ascertain if the generic UK HPR1000 design is compliant from the regulatory perspective to be built within the UK. I reviewed this within each of the technical aspects discussed in Section 4 of my report.

47. The radiological protection safety requirements section provides information on NT 1, 2 and 3 and how the PCSR Chapter 22 (Ref. 3) is structured to meet these targets, the scope of the chapter and a route map for the claims, arguments and evidence.

48. The source term section describes the definition of the generic UK HPR1000 design radioactive sources as well as the various source terms for normal operation. There are several source term categories discussed, providing information on definition, radionuclide selection and how they are derived as well as the ALARP demonstration. This is discussed in greater detail within Section 4.2.

49. The RP provided a high-level overview of the ALARP assessment for radiological protection. Information is provided for each step to demonstrate the generic UK HPR1000 design from a radiological protection perspective is ALARP, specifically for external and internal doses for workers during normal operation. This is discussed for each technical aspect within Section 4 of my report. An overview of the demonstration of ALARP is provided in Section 4.10.

50. The RP discussed the radiological protection measures used to control direct radiation and contamination for the generic UK HPR1000 design. This covers numerous topic areas such as designation of areas, access and egress control and radiation shielding for the generic UK HPR1000 design. Radiation and contamination zoning along with access and egress control are discussed within Section 4.6 whilst radiation shielding is discussed in Section 4.3.

51. The RP provided information on the radiation and contamination monitoring design for the generic UK HPR1000 design. The RP provided a description of the functions of the radiation and contamination monitoring system along with the different monitoring channels used for the generic UK HPR1000 design. This aspect is discussed within Section 4.7.

52. The RP provided an overview of collective worker dose for the generic UK HPR1000 design. Within this section the RP provided detailed calculations for collective worker dose, also stating that the doses to workers were ALARP. In addition, the RP provided information on the individual dose to employees working with ionising radiation and dose to other employees on site. This aspect is discussed within Section 4.4.

53. The RP provided an overview for calculating public dose from direct radiation for the generic UK HPR1000 design and provided a demonstration that doses are reduced SFAIRP. An overall dose assessment to members of the public during normal operation is covered by ‘PCER Chapter 7: Radiological Assessment’ (Ref. 15). This aspect is discussed within Section 4.5.

54. The final technical area covered within PCSR Chapter 22 (Ref. 3) is PAA. The RP identified the systems and components to which access is required in post-accident situations and described the methodology for assessing the dose for workers participating in accident mitigation. This aspect is discussed within Section 4.8.

55. PCSR Chapter 22 (Ref. 3) concluded that the radiological protection safety case for the generic UK HPR1000 design, supports the claim that the risk to workers and members of the public from the potential harmful effects of ionising radiation complies with UK legal requirements and is ALARP. The RP process for ALARP demonstration for radiological protection to ensure that exposure to radiation is ALARP, and the radiological protection considerations applied to reactor water chemistry, fluid system design, material selection, equipment design, designation of areas, ventilation, layout

design and radiation shielding are summarised in this chapter. A supporting report 'ALARP Demonstration Report of PCSR Chapter 22' (Ref. 16) presents further evidence in support of these claims.

56. In addition to the above PCSR Chapter 22 (Ref. 3), where required within my assessment I have referenced further PCSR chapters as appropriate.

3.5 Criticality Safety for Fuel Storage Aspects PCSR Chapter 5

57. It should be noted that 'PCSR Chapter 5 Reactor Core' (Ref. 17) provides the criticality safety for fuel storage analysis for fresh fuel and spent fuel storage. This aspect is discussed within Section 4.9.

3.6 Emergency Arrangements PCSR Chapter 32

58. PCSR Chapter 32 (Ref. 18) covers the design information for the emergency preparedness of the UK HPR1000. The emergency arrangements are established to prepare for a radiation emergency and mitigate the consequences in case of an occurrence by taking all reasonably practicable means. PCSR Chapter 32 (Ref. 18) provides sub-claims and arguments to corroborate the claim 3.2.5

"Emergency arrangements will be in place prior to commissioning that will be in accordance with up-to-date standards in the event of a release of radioactive substances".

59. Within PCSR Chapter 32 (Ref. 18) the RP provided details of how emergency arrangements interface with other chapters as well as a high-level summary of the applicable codes and standards which relate to emergency arrangements, for example REPPIR19 (Ref. 11). These have been used by the RP to ascertain if the generic UK HPR1000 design is compliant with current UK standards and guidance.

60. The RP provided details of the emergency management arrangements for the generic UK HPR1000 design as well as a brief overview of the on-site emergency response facilities and the on-site accident management. Further information is provided within Section 4.8.1.6.

3.7 Topic Reports and Supporting Documents

61. As noted above, tier 2 documentation is used to support and substantiate the claims in PCSR Chapter 22 (Ref. 3) and provide the link between that and the further detailed evidence and analysis presented elsewhere in the generic radiological protection safety case. Key tier 2 submissions for the purposes of my assessment are the numerous topic reports covering:

- Source term
- Radiation and contamination zoning
- Radiation shielding
- Radiation and contamination monitoring
- ALARP assessment
- Worker dose evaluation
- Public dose evaluation from direct radiation
- PAA analysis.

62. The topic reports generally present the objectives and justification for Radiological Protection in particular areas of the plant or for a particular process. My assessment of the topic reports is within Section 4.

63. Tier 3 submissions also include a series of supporting documents covering the:

- Nuclear buildings source term analysis
- Radiation shielding design
- Compartment classification and radiation zones
- Radioactive systems source term analysis
- Radiation classification.

64. My assessment of the above submissions is within Section 4.

3.8 Responses to RQs and ROs

65. As a result of my assessment of the generic UK HPR1000 safety case, I raised 3 ROs (Ref. 19) and 78 RQs (Ref. 20). In ‘Production Strategy for Radiological Protection’ (Ref. 13) the RP states that the responses to the RQs and ROs were largely incorporated into the tier 2 or tier 3 documentation of the Radiological Protection and Criticality safety case. I have considered the responses to ROs and RQs as part of my assessment and, as described in Section 4, I have considered the extent to which adequate consolidation of such information into the Radiological Protection and Criticality safety case has been achieved.

4 ONR ASSESSMENT

4.1 Structure of Assessment Undertaken

66. The structure of this section of my Step 4 of GDA report is aligned with that of my assessment scope and sampling strategy described in Section 2. I have considered safety-significant parts of the Radiological Protection and Criticality safety case associated with how the generic UK HPR1000 design will be operated.

67. My Step 4 of GDA report concludes my assessment of the overall demonstration that relevant risks have been reduced ALARP and of the consolidated safety case. I have also provided a summary of the standards, guidance and RGP that I have used at the end. The resulting chapter structure is as follows:

- Normal operation – source terms
- Radiation shielding
- Worker dose assessment
- Public dose assessment
- Radiation and contamination zoning
- Radiation and contamination monitoring
- PAA
- Criticality safety for fuel storage
- Demonstration that relevant risks have been reduced ALARP
- Consolidated safety case
- Comparison with define standards, guidance and RGP.

68. This structure includes all the areas of scope that I targeted for Step 4 of GDA, as per my assessment strategy recorded in Section 2 and the Step 4 of GDA Radiological Protection and Criticality assessment plan (Ref. 5). The assessment plan identified items for follow-up from the Step 3 of GDA (Ref. 21). I have assessed all of the items within the following sections.

4.2 Normal Operation – Source Terms

4.2.1 Assessment

69. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I undertook an assessment of the RP's approach to minimisation of the normal operation source terms. Within PCSR Chapter 22 (Ref. 3), the RP has the following sub-claim:

"The risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP"

70. This sub-claim is part of the overarching claim which encompasses workers and members of the public. The RP provide arguments and evidence to corroborate this claim through PCSR Chapter 22 Section 22.5. The RP describes the definition of the generic UK HPR1000 design radioactive sources and covers the various source terms for normal operation and how they are used for radiological protection purposes. The information provided is a high-level overview of the following (Ref. 3):

- Source term categories
- Radionuclide selection
- Application of OPEX
- Demonstration of ALARP/Best Available Technique (BAT)

71. The scope of my Step 4 assessment covers only the aspects associated with 'normal operation' source terms. 'Accident' source terms were assessed by the Chemistry and

Fault Studies technical disciplines. Source term assessment is a multi-disciplinary topic area, which for the purposes of Step 4 of GDA has been led by Chemistry with input from the Radiological Protection, NLR and the Environment Agency topic areas.

72. Source term for the generic UK HPR1000 design is defined by a suite of documents supporting PCSR Chapter 22 (Ref. 3). The top level of which is the 'Normal Operation Source Term Strategy Report', currently at Rev D (Ref. 22) . This is supported by a number of documents including:

- 'Report of Radionuclide Selection during Normal Operations' (Ref. 23)
- 'Normal Operation Source Term General Report' (Ref. 24)
- 'Primary Coolant Source Term Methodology Report' (Ref. 25)
- 'Primary Coolant Source Term Calculation Report' (Ref. 26)
- 'Airborne Activity Supporting Report' (Ref. 27)
- 'Derived Source Term Supporting Report' (Ref. 28)
- 'Radiological Protection Technical User Source Term Report' (Ref. 29)

73. I assessed the source term reports during Steps 3 and 4 of GDA against RGP and guidance in ONR SAPs (RP.7) (Ref. 2) and NS-TAST-GD-038 (Ref. 4) on general matters such as ALARP. ONR regards RGP as basing the source term on relevant OPEX, amended to the design and safety case specifics supported and supplemented by technical justified calculation where appropriate.

74. The RP took a reasonable approach to source term with a logical documentation structure defined. Significant radionuclides were identified covering corrosion products, activation products, fission products and actinides. Source terms are provided for all normal operation plant conditions including transients.

75. As reported in the Step 3 of GDA Radiological Protection assessment note (Ref. 21), assessment of the topic report on 'Application of Cobalt Based Alloy in SSCs' (Ref. 30), found that the justification provided by the RP that the source term had been minimised was not adequate. The radiochemistry of the primary circuit plays a key role in minimisation of source terms and the Chemistry technical discipline had concluded the RP had only provided very limited evidence to support the claims and arguments that justify the radiochemistry of the primary circuit and that the radioactivity in the primary circuit had not been shown to be minimised. As a result of this, an RO was raised, led by the Chemistry technical discipline, identifying ONR expectations on these matters. This assessment is detailed in section 4.2.1.2.

4.2.1.1 Overview of Source Term Assessment

76. The 'Normal Operation Source Term Strategy Report' (Ref. 22) provides an overview of the different source term categories for the generic UK HPR1000 design.

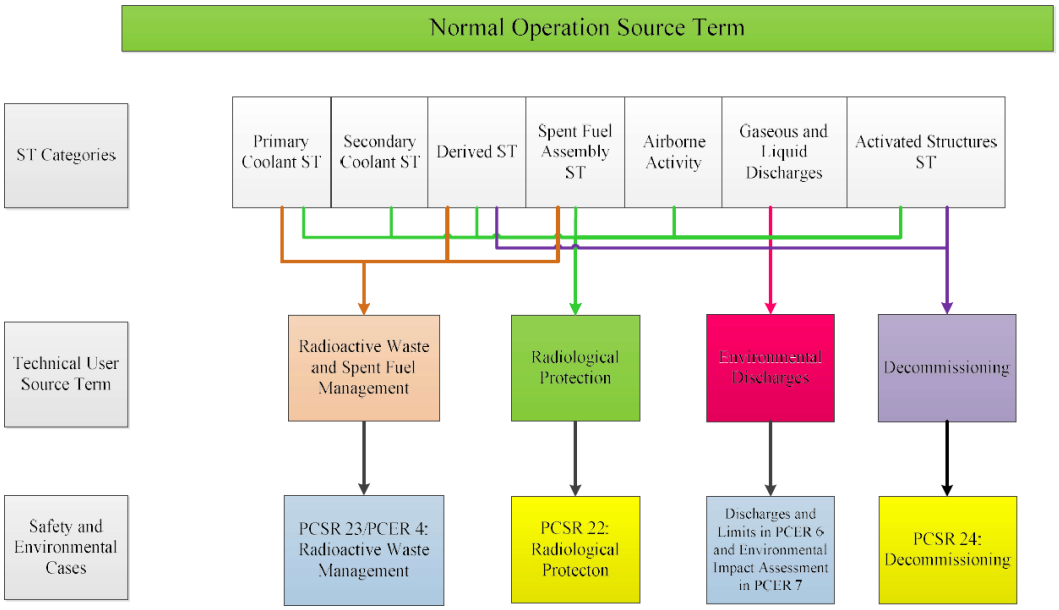

Figure 2: Relationship between the normal operation source term and the technical users source terms (Ref. 22).

77. From figure 2 it can be seen that the RP has an array of source terms which are assessed in various technical disciplines. Regarding radiological protection I assessed the following source terms:

- Primary coolant source term
- Secondary coolant source term
- Derived source term
- Spent fuel assembly source term
- Airborne activity source term
- Activated structures source term

78. The above source terms are discussed in greater detail within the 'Technical User Source Term Report' (Ref. 29). From the review of this document, several clarifications were required, hence I raised RQ-UKHPR1000-999 (Ref. 20). From the RP response they adequately resolved all queries, and I judged the technical user source term report to be adequate.

79. In addition, further queries were raised on the calculation for the 'Airborne Activity Supporting Report' (Ref. 27) as well as the 'Derived Source Term Supporting Report' (Ref. 28). These were raised in RQ-UKHPR1000-1317 and RQ-UKHPR1000-1316 respectively (Ref. 20). From the RP response they adequately resolved all queries, and I judged the information provided on airborne activity and the derived source term to be reasonable.

4.2.1.2 Minimisation of Radioactivity

80. As indicated in section 4.2.1, there was insufficient justification by the RP on how radioactivity had been minimised and that the associated risks had been reduced SFAIRP during Step 3 of GDA. When initially raised with the RP, the response was that this information and justification was already presented in the source term submissions, including the 'Primary Coolant Source Term Calculation Report' (Ref. 31) and the 'Derived Source Term Supporting Report' (Ref. 28).

81. Whilst the source term submissions present an OPEX-based estimate of various nuclides, ONR judged there to be potential regulatory shortfalls associated with the

demonstration that radioactivity had been reduced SFAIRP within the primary circuit and other associated systems of the generic UK HPR1000 design. The extant safety case provided only very limited evidence to support the claims and arguments that had been made about the control of radioactivity. As such, RO-UKHPR1000-0026 (Ref. 19) was raised by the Chemistry technical discipline with support from Radiological Protection and NLR from a waste and decommissioning perspective. The RO required the RP to:

- Quantify and characterise all radioactive species in the primary circuit and connected systems;
- adequately justify the estimates of radioactivity;
- substantiate the systems, controls and measures that will be used to minimise and remove the radionuclides;
- demonstrate that all reasonably practicable measures have been taken to reduce radioactivity;
- provide an adequate quantitative estimate of the inventory and associated activities of the corrosion products expected to be generated and transported in the primary circuit; and
- estimate the quantities of activated corrosion products present in all relevant systems connected to the primary circuit.

82. To resolve RO-UKHPR1000-0026 (Ref. 19) the RP stated they would provide the following:

- Identify the sources of radionuclides and key radionuclides of interest;
- identify the transport mechanisms and the main Structures, Systems and Components (SSCs);
- quantify and characterise, all relevant radioactive species;
- identify the measures that ensure radionuclide generation and transport and radioactivity levels are prevented, minimised and optimised;.
- demonstrate all relevant measures have been considered and implemented for the generic UK HPR1000 design, with due consideration for proportionality;
- estimate and discuss the quantities of activated corrosion products present in the primary circuit and systems connected to the primary circuit; and
- using this information, demonstrate that radioactivity has been reduced SFAIRP.

83. The RP's response to RO-UKHPR1000-0026 (Ref. 19) was to provide two deliverables, 'Minimisation of Radioactivity Route Map Report' (Ref. 32). and 'The Corrosion Product Source Term Analysis with UKHPR1000 Specific Design' (Ref. 33). Chemistry led and coordinated the assessment of these documents with support from Radiological Protection, NLR (Ref. 34) and the Environment Agency.

84. From this assessment, there were a number of shortcomings the chemistry assessor and I identified, hence the following RQs were raised (Ref. 20):

- RQ-UKHPR1000-0995;
- RQ-UKHPR1000-1121; and
- RQ-UKHPR1000-1668.

85. The Radiological Protection assessment queries were included in the Chemistry RQs as appropriate, and included: deposition of corrosion products, justification that radioactivity had been minimised SFAIRP, optioneering and use of Stellite™, use of secondary neutron sources and items to be left for the site licensing stage. From the RP response to the above RQs and discussions over the Step 4 of GDA period with Chemistry, NLR and Radiological Protection, the respective technical disciplines were

content that RO-UKHPR1000-0026 (Ref. 19) could be closed out with the following Chemistry led matters identified (Ref. 34) where further evidence is required:

- Demonstrate that operating practices have been optimised to ensure radioactivity has been reduced SFAIRP for the generic UK HPR1000 design.
- Demonstrate that chemistry practices in start-up and shutdown have been optimised to ensure radioactivity has been reduced SFAIRP for generic UK HPR1000 design.
- Demonstrate that the amount of high-cobalt bearing materials in contact with the primary coolant has been reduced SFAIRP for generic UK HPR1000 design.
- Demonstrate that the systems supporting the generic UK HPR1000 design for primary circuit, (such as the Chemical Volume and Control System (RCV[CVCS]), are adequately substantiated against their design requirements.
- Quantify the generation of corrosion products from materials containing high amounts of cobalt in the primary circuit for the generic UK HPR1000 design.
- Adequately underpin the Silver-110 (^{110m}Ag), Antimony - 122 (^{122}Sb) and Antimony - 124 (^{124}Sb) source term for the generic UK HPR1000 design.

86. The above matters, all of which potentially relate to and could impact Radiological Protection, were characterised as part of the Step 4 of GDA report for Chemistry. The following Assessment Findings (AFs) have been raised within the Chemistry Step 4 of GDA report (Ref. 35):

AF-UKHPR1000-0112 – The licensee shall, as part of detailed design, justify that the cobalt inventory within UK HPR1000 has been optimised, taking account of the relevant risks. The justification should include the impacts of different operating conditions and material choices on radioactivity, worker doses and component reliability.

AF-UKHPR1000-0113 – The licensee shall, as part of detailed design, justify which valves will include cobalt-based hard facings. This should focus on those which do not fall into the “prohibited” category according to procurement specifications. The justification should demonstrate that relevant risks have been reduced so far as is reasonably practicable.

AF-UKHPR1000-0114 – The licensee shall, as part of detailed design, demonstrate that the manufacturing routes of key structures systems and components reduce radioactivity so far as is reasonably practicable.

AF-UKHPR1000-0115 - The licensee shall, as part of developing the operational chemistry safety case, demonstrate how operating practices have been optimised to reduce the risks associated with the generation, transport and accumulation of radioactivity, so far as is reasonably practicable. This should include the activities necessary during start-up and shutdown of the plant.

AF-UKHPR1000-0117 – The licensee shall develop the plant-specific corrosion product estimations, including all relevant factors, to justify that risks have been reduced so far as is reasonably practicable. The analysis should assess how the coolant chemistry and the plant design and operation, affect the generation and transport of all relevant corrosion products.

AF-UKHPR1000-0118 – The licensee shall, as part of developing the operational safety case, define limits and conditions to ensure that the risks associated with tritium (3H) have been reduced so far as is reasonably practicable.

87. Notwithstanding the above shortfalls raised, in my judgement, and from my perspective for the purposes of Step 4 of GDA, I was satisfied that the RP has:

- Improved the methods of minimising and controlling radioactivity for the purposes of normal operations source terms;
- provided a conservative value on its optimisation; and
- provided sufficient evidence to close out RO-UKHPR1000-0026 (Ref. 19).

88. It should be noted Radiological Protection will be involved in the resolution of the Chemistry led AFs as a related technical topic area.

4.2.1.3 OPEX

89. To generate the normal operation source terms, the RP has mostly used OPEX from the existing fleet of reactors in China that are operated by CGN. In consequence, this information has been produced from nuclear reactors that are similar, but not identical to the generic UK HPR1000 design. Where OPEX is not available, the source term is calculated using theoretical calculations based on radionuclide production and decay. Two values are calculated for the source term (Ref. 24):

- Realistic values – representing a best estimate of the activity concentration of radionuclides expected in the generic UK HPR1000 design during normal operation.
- Design values – representing a conservative maximum for the source terms which can be considered a bounding limit for the plant design. It is anticipated that this limit would not be exceeded during operation and includes events which are expected to occur in the lifetime of the plant.

90. As reported in the Step 3 of GDA Radiological Protection assessment note, (Ref. 21) the information in the topic reports at that time indicated that further underpinning OPEX was required. In consequence, and as part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I assessed the general adequacy of the normal operation source terms through the following documents:

- ‘Report of Radionuclide Selection during Normal Operation’ (Ref. 23)
- ‘Primary Coolant Source Term Methodology Report’ (Ref. 25)
- ‘Primary Coolant Source Term Calculation Report’ (Ref. 31)

91. I raised six RQs to facilitate my assessment, and I made use of three RQs raised by Chemistry (Ref. 20):

- RQ-UKHPR-0690 (Chemistry)
- RQ-UKHPR-0691 (Chemistry)
- RQ-UKHPR-0692 (Chemistry)
- RQ-UKHPR-0712
- RQ-UKHPR-0713
- RQ-UKHPR-0714
- RQ-UKHPR-0715
- RQ-UKHPR-0717
- RQ-UKHPR-1405.

92. From the above assessment I produced an assessment note, (‘Assessment of Source Term Documentation’ (Ref. 36)). This focussed upon the choice of radionuclides, the

use of OPEX and calculations, and the magnitude of the source term that was generated. Comparisons were made between the source term for the generic UK HPR1000 design and other similar designs, and it was concluded that the radionuclide activity concentrations calculated for the generic UK HPR1000 design source term are reasonable for the calculation of both realistic and design values. However, the review identified a number of issues in terms of how OPEX had been used by the RP to develop the generic UK HPR1000 design source term. The assessment found the following:

- Not all of the CGN OPEX referenced by the RP provided sufficient explanation for the selection of OPEX from the datasets. The information as presented did not demonstrate that the RP has presented CGN OPEX in a systematic way, I consider this to be a minor shortfall.
- The RP had not adequately demonstrated that it had a systematic method for selecting non-CGN OPEX; I consider this to be a minor shortfall. Such a method would have set out the totality of the relevant OPEX available to the RP and the logical steps by which this is selected and utilised in the Generic UK HPR1000 safety case.
- The RP had not presented adequate detail on or explanation of the EDF OPEX employed in its source term case, I consider this to be a minor shortfall.
- The RP's case for the radionuclides retained in the generic UK HPR1000 design for primary coolant source term is adequate for all radionuclides other than Iron-55 (^{55}Fe). I consider this to be a minor shortfall due to the low radiological protection significance of ^{55}Fe.
- The RP had not initially substantiated the parameter and input values for the calculation methods it had employed. It should be noted that following the subsequent response to RQ-UKHPR1000-1405 (Ref. 20), I concluded that the RP had adequately substantiated the parameter values used in the calculation methods. I also concluded that the PALM V1.4.2 code was adequate for its role in the calculation of the fission product and actinide normal operation source terms.

93. From my assessment, ('Assessment of Source Term Documentation' (Ref. 36)), I was in general, content with the RP's substantiation of the source term. I did find however, that the RP substantiation of the radionuclide selection for the deposited corrosion product source term was unclear. I was also uncertain in the submissions and RQ responses how the deposited corrosion product source term was used in the safety case. The assessment found the following:

- The radionuclide Cobalt-60 (^{60}Co) has not been evaluated in the steam generator (only Cobalt- 58 (^{58}Co) is), despite evidence that it could be significant and detectable.
- The RP did not demonstrate that the aggregated impact of the non-retained radionuclides at the limit of detection – Manganese - 54 (^{54}Mn), Iron - 59 (^{59}Fe), ^{110m}Ag and, in the case of the steam generator ^{124}Sb - would not materially affect a calculation of total gamma activity sufficiently to impact on the radiological protection safety case (which would justify their non-inclusion).
- The RP has based the radiological protection design on measurements of the total gamma activity derived directly from CGN OPEX.

94. From my assessment in the preceding paragraph, my judgement is that the RP could have provided further explanation on how the deposited corrosion product source term is used in the safety case. The total gamma activity has been used as a surrogate, which could give rise to additional radiation exposure in some areas of the generic UK HPR1000 design if the corrosion product source term is applied. In my judgement this does not impact my overall decision that the source term is broadly acceptable, though I consider this a finding rather than a shortfall. As such I have raised the following AF:

AF-UKHPR1000-0096 – The licensee shall demonstrate how the deposited corrosion product source term is applied and used within the safety case. Corrosion products are a significant source of operational radiation exposure and the licensee shall optimise this source term to reduce worker dose so far as is reasonably practicable.

95. It should be noted that a related finding has been raised by my Chemistry colleagues, AF-UKHPR1000-0117 within the Chemistry Step 4 of GDA report (Ref. 35).

96. In addition, an RO (RO-UKHPR1000-0044 – 'Identification and Use of OPEX in the UK HPR1000 Generic Design and Safety Case' (Ref. 19)) was drafted in parallel with my Step 4 of GDA Radiological Protection assessment of the source term documentation (Ref. 36). This required the RP to provide further information on identification and use of OPEX in the generic UK HPR1000 safety case. The RO identified gaps in the RP's approach with regard to:

- identifying, justifying and using OPEX;
- scope (i.e. depth and breadth) of the OPEX identified and selected; and
- how the OPEX is presented in the safety case.

97. Radiological Protection was one of the topics in which OPEX is important and was considered as part of the closure of this cross-cutting RO (Ref. 37). It was concluded the overall intent of the RO had been met as the RP had:

- Presented suitable and sufficient evidence to demonstrate a robust process has been applied to identify topics placing the greatest reliance on OPEX to make an adequate demonstrate of safety.
- Taken ownership of gaps and shortfalls in the extant generic safety case, related to OPEX, and undertaken an appropriately targeted review to identify areas requiring improvement, as compared against their revised, strengthened arrangements, for managing OPEX (Ref. 38).
- Made improvements to existing generic safety case documents which utilise OPEX to either: improve referencing and traceability to sources of OPEX; or provide more robust justifications for how relevant OPEX has been screened and selected.
- Produced a radiological protection OPEX summary report (Ref. 39), which is used as key inputs into relevant ALARP demonstrations. These additions to the generic safety case improve the overall visibility of how relevant OPEX has been identified, screened and justified as being applicable. The RP also provided a more robust demonstration of how relevant OPEX has been integrated into the generic safety case.
- From my sampling of the 'Normal Operation Source Term General Report' (Ref. 24) and the 'Primary Coolant Source Term Methodology Report' (Ref. 25) I found that the RP had updated these reports in line with the response to this RO requirements, to provide greater clarity and visibility on OPEX sources, and the application of the OPEX screening and selection process.

98. My sampling of the two updated documents mentioned in the previous paragraph has shown that the RP has provided a more robust and transparent demonstration of how OPEX has been selected and used. Further documents were not sampled during Step 4 of GDA and as such, implementation of the strengthened arrangements for managing OPEX (Ref. 38), across the suite of source term documentation needs to be tracked post Step 4 of GDA. With this in mind it is my judgement that further work needs to be undertaken regarding OPEX. As such I have raised the following AF:

AF-UKHPR1000-0097 – The licensee shall ensure that the operating experience used as evidence for the source term for normal operations is robustly underpinned and documented within the safety case.

4.2.1.4 Generation, Transport and Behaviour of Tritium During Normal Operations

99. The Chemistry technical discipline, in conjunction with Radiological Protection, raised RO-UKHPR1000-0049 (Ref. 19). This looked at the generation, transport and behaviour of ^{3}H during normal operations. As part of the resolution of this RO, I raised RQ-UKHPR1000-1677 (Ref. 20) from the assessment of 'identification and minimisation demonstration of ^{3}H related safety risk' (Ref. 40). This RQ assessed ^{3}H aspects regarding radiation protection and the calculation of internal radiation exposure. From the RP responses they adequately resolved all queries raised in the RQ, and I judged the radiological protection aspects to be resolved regarding RO-UKHPR1000-0049. Further information is available within the Step 4 of GDA Chemistry report (Ref. 35).

4.2.2 Strengths

100. From the above assessment as outlined in Section 4.2, the following strengths have been identified relating to source terms:

- The RP has developed a safety case with an improved demonstration that radioactivity has been reduced SFAIRP.
- The RP's use of summary statistical descriptors for CGN OPEX data sets is an effective way of providing adequate quantitative information where measured data is confidential.
- The RP has been able to obtain non-CGN OPEX.
- The RP has generated a list of retained radionuclides that agree well with the lists found in a comprehensive range of information sources, including those found in reactors of similar design that have already undergone the GDA process.
- The theoretical calculation methodologies employed by the RP are reasonable.
- The calculated generic UK HPR1000 design source term agrees well with multiple comparators.
- The RP has provided sufficient evidence to corroborate their claims and arguments in relations to the source terms for the generic UK HPR1000 design form a radiological protection perspective.

4.2.3 Outcomes

101. Based on my assessment, I was satisfied that the RP calculations for radionuclide activity concentrations for the generic UK HPR1000 design for source term, have been improved, for both realistic and design values.

102. From reviewing the information, the RP has provided sufficient evidence to corroborate the claim made within this section.

103. Due to the work undertaken for the resolution of RO-UKHPR1000-0026 (Ref. 19), the RP has demonstrated that the risk to workers from the potential harmful effects of ionising radiation due to the normal operation source term has been reduced during the course of GDA, although further work is required to fully demonstrate that this has been minimised SFAIRP.

104. Radiological protection aspects for RO-UKHPR1000-0026, RO-UKHPR1000-0044 and RO-UKHPR1000-0049 (Ref. 19), have been sufficiently resolved and closed.

105. I am also satisfied that overall, the RP has employed improved methods to assess normal operation source terms.

106. However, my assessment has identified the following AFs:

- AF-UKHPR1000-0096 – The licensee shall demonstrate how the deposited corrosion product source term is applied and used within the safety case. Corrosion products are a significant source of operational radiation exposure and the licensee shall optimise this source term to reduce worker dose so far as is reasonably practicable.
- AF-UKHPR1000-0097 – The licensee shall ensure that the operating experience used as evidence for the source term for normal operations is robustly underpinned and documented within the safety case.

107. I have identified four minor shortfalls as discussed in Section 4.2.

4.2.4 Conclusion

108. Based on the outcome of my Step 4 of GDA of source terms, I have concluded that the claims, arguments and evidence are reasonable, and that the RP has produced an adequate safety case for the purposes of Step 4 of GDA that fulfils the sub-claim stipulated with the PCSR Chapter 22 (Ref. 3).

109. Whilst I have identified two AFs in my assessment, it is my opinion that these can be resolved by the RP during site-specific stages.

4.3 Radiation Shielding

4.3.1 Assessment

110. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I have undertaken an assessment of the RP approach to radiation shielding design to determine that the use of radiation shielding reduces occupational and public exposures SFAIRP.

111. Within the PCSR Chapter 22 (Ref. 3), the RP identifies radiation shielding as one of the radiological protection measures used to support "Claim 3.4.4: The risk to workers and members of the public from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP."

112. Although in the UK there is no specific legislation governing the requirements and acceptability of radiation shielding, the utilisation of effective radiation 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 within IRR17 (Ref. 10). As a result, I have considered radiation shielding design to be a principal aspect of my assessment.

113. The radiation shielding assessment was undertaken to assess the generic UK HPR1000 design for radiation shielding provisions identified in the PCSR Chapter 22 submission (Ref. 3), to review the arguments presented in the PCSR Chapter 22, and to assess whether the evidence presented substantiated those arguments for radiation shielding. The objectives of the assessment were as follows:

- To be satisfied that the generic UK HPR1000 design for radiation shielding design fulfilled the expectations outlined in the SAPs (Ref. 2), in particular RP.6 and in the NS-TAST-GD-002 for radiation shielding (Ref. 4).
- To be satisfied that RGP had been applied to the radiation 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. 41).

114. I was assisted in my assessment by ONR's specialist TSC (Ref. 42). The TSC initially undertook reviews of the key radiation shielding documents submitted by the RP.

Based on the findings of this initial review, the TSC and I raised RQs requesting further information and documents for subsequent review. These RQs included requests for detailed information for five radiation shielding assessment samples selected by the TSC for “deep dive” review. A RO was also raised as a result of some of the findings (RO-UKHPR1000-0060 (Ref. 19)); details of which are provided in later sections. I undertook a final review of the latest revisions of the key radiation shielding documents which had been updated based on the findings of the RQs and RO.

115. The key radiation shielding documents that were provided to support the safety case for radiation shielding provision are listed below:

- PCSR Chapter 22 (Ref. 3);
- radiation shielding methodology reports (Ref. 43, Ref. 44, Ref. 45, Ref. 46);
- radiation Shielding Design Reports (SDRs) (Ref. 47, Ref. 48, Ref. 49, Ref. 50); and
- detail radiation shielding assessment reports (Ref. 51, Ref. 52, Ref. 53, Ref. 54, Ref. 55).

116. Additional information and supporting documentation were also sampled and are noted where relevant in the sections below.

117. The technical review of the above documentation and additional information requested in RQs focussed on the following areas:

- Source Terms: High-level review of the source terms and their use in radiation shielding assessments.
- Radiation shielding Design Basis Data**:** Review of physical data (e.g. material densities and compositions, flux to Dose Conversion Factors (DCFs)) used as the basis for calculations.
- Calculation Methods: Validation of computational codes used in the generation of source terms and the calculation of dose rates to support radiation shielding assessments.
- Review of Radiation shielding Assessments: Review of the RP’s methods, data and resulting radiation shielding provisions and dose rates; and
- Dose Uptake and ALARP: A review of the public and worker dose uptake from direct radiation exposures and demonstration that they are ALARP.

118. The following sections summarise the findings of the radiation shielding assessment review. Further details of the TSC findings can be found in their report (Ref. 42).

4.3.1.1 Source Terms

119. A general review of the source term documentation is within Section 4.2 of this assessment. The TSC has also undertaken sampling of the source terms with regards their suitability and use in radiation shielding assessments. Their review (Ref. 42) sampled the following reports:

- ‘Radiological Protection Technical User Source Term Report’ (Ref. 29)
- ‘Primary Coolant Source Term Calculation Report’ (Ref. 31)
- ‘Secondary Coolant Source Term Supporting Report’ (Ref. 56)
- ‘Derived Source Term Supporting Report’ (Ref. 57)
- ‘Fuel Building Source Term Analysis Report’ (Ref. 58)
- ‘Spent Fuel Assembly Source Term Supporting Report’ (Ref. 59)
- ‘Activated Structures Source Term Supporting Report’ (Ref. 60)

120. No significant shortfalls were identified, and it was noted that the RP states that ‘design’ source terms were used for radiation shielding design, whereas ‘realistic’ source terms were used for worker and public dose assessments. The TSC considered

this to be a reasonable approach and noted the design values are generally significantly higher than the realistic values, though in some cases the values are identical.

121. Concerns were raised with regards to the use of source terms in calculations for two of the radiation shielding assessment samples (see Section 4.3.1.4). There were errors in the treatment of radionuclide source terms which had been incorrectly decayed and did not appropriately take into account the production of radioactive daughter products. It should be noted that these discrepancies in the source terms for these samples did not significantly impact the conclusions of the assessments and the RP took immediate action to remedy these issues. Whilst these discrepancies regarding the source terms may not have significantly impacted the dose rates for the radiation shielding assessment samples reviewed by the TSC, this may not be the case if they are present in other radiation shielding assessments for the generic UK HPR1000 design. It is on this basis that I have raised the following AF:

AF-UKHPR1000-0098 – The licensee shall demonstrate that the source terms used in radiation shielding assessments are justified and robust. This should address the shortfalls identified during Step 4 of GDA, including but not limited to, ensuring that radionuclide source terms for shielding assessments are correctly decayed and incorporate equilibrium daughter products, where appropriate.

4.3.1.2 Radiation Shielding Design Basis Data

122. A review of the physical properties of the radiation shielding materials used for the generic UK HPR1000 design radiation shielding calculations was undertaken through sampling of the 'Radiation Shielding Topic Report' (Ref. 43) and by raising RQ-UKHPR1000-0990 (Ref. 20). The radiation shielding material compositions and densities used by the RP are typical of those used within the UK nuclear industry.

123. Radiation shielding calculations inconsistently used outdated DCFs when calculating dose rates through radiation shielding (Ref. 61). Calculations performed by the RP used default DCFs within the MicroShield version 7.02 and SuperMC calculation codes which are derived from International Commission on Radiological Protection (ICRP) publications ICRP51 (Ref. 62) and ICRP21 (Ref. 63) respectively. I consider this to be a minor shortfall, as it is RGP to use the most recent DCFs recommended by ICRP, which in this instance are those outlined in ICRP116 (Ref. 64). The RP subsequently provided satisfactory evidence in response to RQ-UKHPR1000-1013 (Ref. 20) to demonstrate that dose rate estimates using the outdated DCFs will be slightly more conservative than those estimated using the current ICRP116 DCFs and fit-for-purpose.

4.3.1.3 Calculation Methods

124. The RP has employed the use of computational codes to facilitate the generation of source terms and the calculation of dose rates to support radiation shielding assessments. As the majority of the codes used by the RP are not typically used within the UK nuclear industry, the TSC has undertaken reviews of the verification and validation (V&V) for the following computational codes (Ref. 42), taking into account guidance provided (Ref. 65, Ref. 66, Ref. 67).

125. PALM is used to undertake depletion and source term calculations, the results of which feed into radiation shielding assessments for the generic UK HPR1000 design. The TSC reviewed the RP's submissions (Ref. 68, Ref. 69, Ref. 70) and information provided in response to RQ-UKHPR1000-1215 (Ref. 20). The review considered the V&V evidence, the quality assurance of the code, how sensitive the results are to user-defined parameters and the expected uncertainties. The information provided by the RP was generally considered to be satisfactory. Some minor matters were identified by

the TSC with regards to how the uncertainties in neutron fluxes affect uncertainties in the activation calculations, a deficiency in the explanation of discrepancies between PALM results and those of other codes, and the management of software bugs. I consider these to be three minor shortfalls with regards to the information provided by the RP to fully demonstrate RGP and that there is no significant reason why this code cannot be considered adequate for radiation shielding assessments.

126. MicroShield version 7.02 has been used for most of the calculated dose rates reported in the SDRs. This code is widely used for similar calculations throughout the nuclear industry and validation support is provided by the vendor. The RP's use of standard material definitions has ensured that build-up and attenuation factors are within the ranges of validity of the data. It is noted that the RP is using a rather old version of the code (Version 12 being the most recent); this will not significantly impact results of the radiation shielding assessments.

127. SuperMC is a Monte Carlo code which has been used for a limited number of calculations in the radiation shielding design. It has typically been used for calculations where the geometry is more complex, such as labyrinths. The TSC reviewed the RP's submissions (Ref. 71, Ref. 72) and information provided in response to RQ-UKHPR1000-1212 (Ref. 20). In general, the TSC considered the evidence provided by the RP to be reasonable, although there was a lack of evidence for photon dose rates. This concern is mitigated to some extent by the TSC's independent calculations (see paragraphs 137 & 138 below), which show reasonable agreement between MCBEND and SuperMC. There were minor matters identified by the TSC with regards to a lack of consideration of uncertainties in using the code and a lack of evidence regarding cross-checking of any SuperMC calculations either in the 'SuperMC Code Verification and Validation Report' (Ref. 71) or the SDRs. I consider these to be two minor shortfalls with regards to the information provided by the RP to fully demonstrate RGP and that there is no significant reason why this code cannot be considered adequate for radiation shielding assessments.

128. JMCT is a Monte Carlo code used by the RP to predict neutron fluences at the RPV inner surface and quarter thickness (T/4) locations, and at surveillance capsule locations so lead factors can be calculated. This is presented in the 'Primary Shielding Calculations Report' (Ref. 44). The TSC reviewed the RP's submission (Ref. 73) and information provided in response to RQ-UKHPR1000-0919 (Ref. 20). The RP responses were sufficient to consider JMCT suitably validated for its intended purpose of RPV neutron fluence assessments.

129. JSNT has been used to calculate both neutron and photon fluxes within the generic UK HPR1000 design for the reactor and dose rates up to the outside of its primary concrete shield during at-power operation; the results of these calculations are reported in 'Primary Shielding Calculations Report' (Ref. 44). To the best of the TSC's knowledge this code has not been used for assessing dose rates beyond the immediate vicinity of the reactor and so has not directly fed into the determination of doses to the workforce or to the public since the reactor is not accessible during power operation. The TSC reviewed the RP's submission (Ref. 74) and information provided in response to RQ-UKHPR1000-0919 (Ref. 20). The TSC concluded that the V&V report for the use of JSNT for PWR radiation shielding calculations was adequate as re-issue of the V&V report resulted in significant improvements. There was a minor matter with regards to the lack of discussion about uncertainties in the calculations which is not consistent with good practice. I consider this to be a minor shortfall with regards to the information provided by the RP to fully demonstrate RGP and that there is no significant reason why this code cannot be considered adequate for radiation shielding assessments.

130. In summary, the RP responded well to the TSC's initial comments on the V&V of the computer codes used and was willing to perform and report additional work to improve the V&V reports. Based on the TSC's review (Ref. 42), I am therefore satisfied that the RP has provided sufficient evidence with regards to V&V to support their use in radiation shielding assessments. It should be noted that the TSC has identified several matters regarding the RP's lack of discussion and treatment of uncertainties for computational codes when considering guidance (Ref. 65, Ref. 66, Ref. 67). Although I have raised six minor shortfalls with regards to the information provided by the RP to fully demonstrate RGP there is no significant reason why these codes cannot be considered adequate for radiation shielding assessments.

4.3.1.4 Review of Radiation Shielding Assessments

131. The SDRs submitted at the beginning of Step 4 of GDA were incomplete with regards to sufficiently covering all of the compartments requiring radiation shielding assessments. Later revisions of the SDRs included additional compartments but also significant changes to previously assessed compartments (including increases in dose rates, radiation shielding provisions and radiation zone classifications). Furthermore, it became apparent that the RP will not undertake any penetration shielding assessments for the UK HPR1000 until after Step 4 of GDA (i.e. during site-specific stages). Based on these initial findings the RP was requested in response to Regulatory Observation Action (ROA)1 of RO-UKHPR1000-0060 (Ref. 19) "to clearly identify and justify what radiation shielding assessments i.e. bulk shielding, openings, penetrations and local shielding, for the UK HPR1000 generic design, have already been completed and those planned to be completed both during, and post Step 4 of GDA".

132. My assessment of the RP's response to this ROA sampled the new report submitted by the RP, 'Radiation Shielding Assessment Strategy and Scope for the UK HPR1000 Generic Design' (Ref. 75) and updates to the methodology reports (Ref. 43, Ref. 45, Ref. 46) and the SDRs (Ref. 47, Ref. 48, Ref. 49, Ref. 50). The information provided by the RP was sufficient to resolve ROA1. The consolidated findings from the initial TSC report (Ref. 42) and my subsequent RO-UKHPR1000-0060 assessment (Ref. 19) are summarised below:

- The RP has provided sufficient clarification in 'Radiation Shielding Assessment Strategy and Scope for the UK HPR1000 Generic Design' (Ref. 75) that all aspects of radiation shielding assessment except for the penetrations shielding have been completed during Step 4 of GDA. It is also evident from my sampling of the latest SDRs that sufficient consideration has been given to bulk shielding, openings, local shielding, transient source shielding and temporary shielding and where necessary assessments appear to have been undertaken.
- Based on my sampling, the SDRs appear to be adequately complete, and the appendices contain sufficient information with regards to summarising the compartments, radiation shielding provisions, dose rates and relevant radiation classification criteria. The appendices would have benefited from including references to the detailed radiation shielding assessments from which this information has been taken for better traceability; I consider this to be a minor shortfall.
- The final revisions of the SDRs have included more discussion of the ALARP process and provided evidence the process has been implemented. In some cases, this has resulted in design changes which aim to lower dose rates.
- The SDRs identify compartments where the dose rates exceed the radiation classification criteria. In cases where additional radiation shielding is not considered practical (and occupancy is expected to be low), the RP has advised that the radiation classification should be increased in line with the estimated dose rates. A matter was identified from the closure of RO-

UKHPR1000-0060 (Ref. 19) with regards to the radiation zoning layouts having not been updated to reflect these changes to the radiation classifications. I consider this to be a minor shortfall.

- I consider deferring detailed assessment of the penetrations until after Step 4 of GDA to be justifiable on the provision that the RP has suitable methods for assessing penetrations, and that there is confidence that the dose rates from penetrations will not significantly impact worker doses or radiation zoning.

133. The **'**Penetrations Shielding Design Report' (Ref. 46) presents the methodology for assessing penetrations throughout the radiation shielding design of the generic UK HPR1000 design. It is important to note that the RP has not commenced assessing penetration shielding for the generic UK HPR1000 design and has also made significant changes to their methodology and underpinning assumptions for assessing penetrations shielding during Step 4 of GDA. As penetration shielding assessments will take place during site-specific stages, there is a need to ensure that the RP's approach represents good practice, is fit-for-purpose and will not significantly impact worker doses or radiation zoning. The RP was therefore requested in response to ROA2 of RO-UKHPR1000-0060 (Ref. 19) to "clearly document the method(s) for identifying and assessing radiation shielding penetrations in the UK HPR1000 generic design and justify why they represent good practice and are fit-for-purpose".

134. My assessment of the RP's response to the ROA sampled the 'Penetration Shielding Design Report' (Ref. 46) and responses to RQ-UKHPR1000-1409 and RQ-UKHPR1000-1748 (Ref. 20). My sampling focussed on ensuring that the RP's approach represents good practice, is fit-for-purpose and will not significantly impact worker doses or radiation zoning. The information provided by the RP was sufficient to resolve ROA2. The consolidated findings from the initial TSC report (Ref. 42) and my subsequent RO-UKHPR1000-0060 assessment (Ref. 19) are summarised below:

- Overall, the latest report (Ref. 46) provides clarity regarding the penetration shielding assessment process and methods previously used for the HPR1000 (Fangchenggang Unit 3) and how these have been improved for the generic UK HPR1000 design. The general principles, processes and methods align with UK RGP and guidance provided in NS-TAST-GD-002 (Ref. 4).
- The report provides additional details and an example of the penetration assessment process previously deployed for the HPR1000 (Fangchenggang Unit 3). This provides some confidence that the penetration shielding principles used for the HPR1000 (Fangchenggang Unit 3) can result in conservative dose rate estimates compared to more detailed analysis using SuperMC calculations.
- There remains a matter not resolved during Step 4 of GDA as the RP has not provided sufficient details as to how the simplified source-shield geometries were chosen or how conservative they are for penetration assessment purposes. It should also be noted that the RP uses different assumptions (e.g. source and compartment dimensions) for assessing penetrations (Ref. 46) and openings (Ref. 45) and has not provided a satisfactory justification as to why this is this is the case.
- Whilst the RP recognises UK RGP with regards to joggled penetrations and shield plates, there is also a matter not resolved during Step 4 of GDA that the RP has not used these radiation shielding measures on the HPR1000 (Fangchenggang Unit 3) and has not provided evidence or examples that they will use them within the generic UK HPR1000 design.

135. In summary, the RP provided sufficient information to close RO-UKHPR1000-0060. Given the above matters, it is my judgement that the RP has not undertaken penetration shielding assessments for the generic UK HPR1000 design and has not provided sufficient evidence to fully demonstrate that their methodologies and

underpinning assumptions are sufficiently conservative to ensure dose rates for all penetrations will be acceptable. It is on this basis that I have raised the following AF:

AF-UKHPR1000-0099 - The licensee shall, during detailed design, demonstrate that all measures have been implemented to reduce doses to workers from radiation shielding penetrations, so far as is reasonably practicable, and ensure the associated effects on radiation zoning are minimised.

136. The 'Radiological Protection Design Principles for Openings' report (Ref. 45) presents the methodology for assessing "openings", by which it means doorways, throughout the radiation shielding design of the generic UK HPR1000 design. The TSC's initial review of the report raised RQ-UKHPR1000-1498 (Ref. 20). The RP has subsequently updated the report in response to the RQ and technical meeting discussions (Ref. 76). Based on my review of the latest report (Ref. 45), there remains a matter regarding the underlying assumptions (e.g. source and compartment dimensions) used in assessing labyrinths not being demonstrably bounding and are inconsistent with those used in penetration assessments (see paragraph 134 above). The report confirms that labyrinths are the preferred option where space allows and sampling of the SDRs provides examples of detailed assessment and optimisation of labyrinths using SuperMC which is in alignment with RGP. In summary, whilst I have reservations regarding the initial methods for assessing labyrinths outlined in the report (Ref. 45), I consider these matters to be a minor shortfall, as I am satisfied that there is evidence in the SDRs (Ref. 48, Ref. 49) and further clarification from technical meetings (Ref. 76) to show that the RP undertakes suitably detailed assessments and optimisation of openings to confirm that radiation shielding provisions and dose rates are acceptable.

137. Sample calculations were provided by the RP in response to RQs (as detailed below) requesting further detailed information regarding radiation shielding assessments selected from the SDRs. The purpose of my sampling was to improve understanding of the RP processes and methodologies and to compare these with UK RGP. The following five samples were requested in response to RQs of which two samples (samples 2 and 3) were subject to independent calculations by the TSC using their own methods and the UK recognised radiation shielding code MCBEND (version 11A RU1):

- Sample 1 - Reactor primary shielding at the hot leg of the primary coolant circuit in response to RQ-UKHPR1000-0990 (Ref. 20).
- Sample 2 - CVCS Chemical Mixing Station BFX2021ZRM in the BFX in response to RQ-UKHPR1000-0989 (Ref. 20).
- Sample 3 - Corridor BSB1034ZRM in the BSB in response to RQ-UKHPR1000-0988 (Ref. 20).
- Sample 4 - Corridor BFX2944ZRM in the BFX in response to RQ-UKHPR1000-1210 (Ref. 20).
- Sample 5 - Analysis Room BNX0957ZRM in the BNX in response to RQ-UKHPR1000-1211 (Ref. 20).

138. The following provides a summary of the TSC's findings for samples 1-3 and my sampling of samples 4 & 5:

- Matters were identified with regards to poor statistical results in sample 1, which indicates that significant information may have been lost and subsequent downstream dose rate and activation calculations may not be sufficiently accurate.
- The TSC found that there was good agreement between MCBEND and SuperMC calculation results for sample 3, indicating that the code SuperMC is capable of carrying out moderately difficult radiation shielding problems.

- The TSC results were in general agreement with those provided in samples 2 and 3. Notable differences in dose rates appeared to be due to differences in modelling assumptions (e.g. penetrations and a doorway) not code performance.
- In one case, a matter was identified with regards to the RP modelling a doorway (i.e. opening) as a concrete wall. The explanation provided by the RP in response to RQ-UKHPR1000-1263 (Ref. 20) did not justify the modelling approximation used and was not considered good practice.
- As previously noted in Section 4.3.1.1 (above) there were matters such as errors in the treatment of radionuclide source terms which had been incorrectly decayed and did not appropriately take into account the production of radioactive daughter products.
- Samples 4 and 5 provide examples of simple bulk shielding calculations using MicroShield. The approach appears to be sufficiently conservative with respect to using design source terms (e.g. during fuel transfers) and geometry modelling. Further confidence in the results of sample 4 was provided through a favourable comparison with previous Fangchenggang Unit 3 assessments. I considered these to be well presented samples that adequately demonstrate the RP's approach and methods for undertaking simple gamma shielding calculations.

139. In summary, the review of the samples has shown some evidence of good practice but has also highlighted several matters which the RP has taken steps to resolve during Step 4 of GDA. Whilst these matters may not have significantly impacted the dose rates for these samples, similar discrepancies in other assessments may have a more significant impact on dose rates and radiation zoning. It is also noteworthy that several matters have been identified in a relatively small selection of assessments. It is my judgement that these matters (highlighted in paragraph 138) are indicative of a shortfall regarding Suitably Qualified and Experienced Personnel (SQEP) and quality management systems for radiation shielding assessments during Step 4 of GDA. It is on this basis that I have raised the following AF:

AF-UKHPR1000-0100 – The licensee shall review and complete its radiation shielding assessments, extending these to the totality of the detailed design and addressing the shortfalls identified during Step 4 of GDA. This should include, but is not limited to, addressing shortfalls regarding modelling assumptions supporting radiation shielding calculations and the statistical accuracy of results.

140. It should be noted that the above AF and related shortfalls are specifically associated with the worker exposure within the sampled areas of the generic UK HPR1000 design. They are not related to assessment of public dose which is discussed within Section 4.5.

4.3.1.5 Dose Uptake from Direct Radiation Exposure & Demonstration of ALARP

141. The dose uptake to the public and workers during normal plant operation will be almost entirely due to direct radiation exposure, for which radiation shielding plays a significant part in ensuring doses are ALARP. The TSC has undertaken a high-level review of the RP's submissions (including RQ responses) with regards to direct radiation exposures and demonstration of ALARP. The TSC assessment considered guidance provided in:

- NS-TAST-GD-038 'Radiological Protection' (Ref. 77),
- NS-TAST-GD-002 'Radiation Shielding' (Ref. 65)
- NS-TAST-GD-043 'Radiological Analysis – Normal Operation' (Ref. 78),
- ONR SAPs (Ref. 2)
- IRR17 (Ref. 10).

142. The findings of the TSC review (Ref. 42) of the 'Worker Dose Evaluation Topic Report' (Ref. 79) and responses to RQ-UKHPR1000-1280 (Ref. 20) are as follows:

- The report presents collective, maximum average and maximum individual worker doses and maximum individual undesignated area worker dose.
- While there is no target set for the collective worker dose it is broadly comparable with other reactor designs.
- The maximum and average individual worker doses are under their respective Basic Safety Limits (BSLs) but higher than the Basic Safety Objectives (BSO).
- The maximum assessed individual dose to a worker in an undesignated area is significantly below the BSO.

143. Similarly, the findings from the TSC review (Ref. 42) of the 'Public Dose Evaluation from Direct Radiation Topic Report' (Ref. 80) and responses to RQ-UKHPR1000-0850 (Ref. 20) are as follows:

- The direct radiation doses to the public are below the BSO.
- The supporting ALARP argument is high-level and qualitative.
- A full ALARP argument which demonstrates adequate weighting of health and safety aspects of public dose should be presented during site-specific stages which includes the BQF and Interim Storage Facility for Intermediate Level Waste (BQZ) and the position of the site boundary once it is confirmed.

144. The TSC assessment of the RP's demonstration of ALARP considers that the documentation provided demonstrates that the RP is fully cognisant of the relevant UK legislation and regulations. The methodology is generally well described and understood and latterly more effort has been made to demonstrate the process more fully. However, the ALARP optioneering process has been target driven with the aim of achieving the dose rate limits rather than a drive to reduce dose rates in general.

145. For worker doses the ALARP process is focussed on high dose rate activities. Source term reduction is the main method cited for driving down lower dose rate activities. For individual doses the RP ALARP justification is that the general trend of dose reduction has continued over the years and that doses are below the BSLs. However, there is a lack of discussion regarding optioneering processes to reduce these doses, both by design, such as reducing occupancy of high dose rate areas or applying more radiation shielding, and by dose efficient procedures. It is recognised that more recent SDRs have provided more evidence of radiation shielding design changes that are again target driven.

146. As stated in paragraph 143, the public dose ALARP demonstration is a basic high-level review which does not yet include all the buildings and only has a provisional and conservative site boundary.

147. A more detailed analysis of the worker and public dose uptake and demonstration of ALARP is also provided in Sections 4.4 & 4.5 respectively.

4.3.2 Strengths

148. From the above assessment as outlined in Section 4.3, the following strengths have been identified relating to radiation shielding:

- The TSC was broadly satisfied that the primary coolant source terms were reasonable.
- Public doses from direct radiation are broadly comparable with other reactor designs and are below the BSO.
- Worker doses from direct radiation, both collective and maximum individual doses, are comparable with other reactor designs. The maximum and average

individual worker doses are below their respective BSLs, and the maximum assessed individual dose to a worker in an undesignated area is significantly under the BSO.
- The RP responded well to RQs and was willing to perform and report additional work which has led to significant improvements in the safety case documents.
- Later versions of the SDRs included more discussion of the ALARP process and provided evidence of the process having been implemented.

4.3.3 Outcomes

149. Based on my assessment, I am satisfied that overall, the RP has employed suitable methods to assess dose rates and specify radiation shielding provisions which will ensure that the public and worker dose from external radiation will be acceptable with respect to standards and guidance outlined in the IRR17 (Ref. 10) and SAPs (Ref. 2).

150. It is evident that the RP has undertaken sufficient work with regards to the assessment of the bulk shielding, openings, local shielding, transient source shielding and temporary shielding for the generic UK HPR1000 design for the purposes of Step 4 of GDA.

151. However, my assessment of the radiation shielding provisions has identified the following AFs.

- AF-UKHPR1000-0098**:** The licensee shall demonstrate that the source terms used in radiation shielding assessments are justified and robust. This should address the shortfalls identified during Step 4 of GDA, including but not limited to, ensuring that radionuclide source terms for shielding assessments are correctly decayed and incorporate equilibrium daughter products, where appropriate.
- AF-UKHPR1000-0099: The licensee shall, during detailed design, demonstrate that all measures have been implemented to reduce doses to workers from radiation shielding penetrations, so far as is reasonably practicable, and ensure the associated effects on radiation zoning are minimised.
- AF-UKHPR1000-0100: The licensee shall review and complete its radiation shielding assessments, extending these to the totality of the detailed design and addressing the shortfalls identified during Step 4 of GDA. This should include, but is not limited to, addressing shortfalls regarding modelling assumptions supporting radiation shielding calculations and the statistical accuracy of results.

152. I also identified a number matters with respect to radiation shielding, many of which were speedily and satisfactorily resolved by the RP. Any remaining matters were detailed as the nine minor shortfalls detailed in Section 4.3.1 (above).

4.3.4 Conclusion

153. Overall, the standard of the final documentation received in support of Step 4 of GDA has been adequate to allow a sufficiently detailed examination of the generic UK HPR1000 design, regarding radiation shielding design and some aspects of detailed radiation shielding provisions.

154. The documentation submitted by the RP in support of the generic UK HPR1000 safety case regarding radiation shielding design demonstrates that, when reviewed in the context of the guidance and expectations outlined in the SAPs (Ref. 2) and NS-TAST-GD-002 (Ref. 4) the radiation shielding provisions are broadly acceptable.

155. Whilst I have identified three AFs in my radiation shielding assessment, it is my opinion that they can be resolved by the RP during site-specific stages and the radiation

shielding design of the generic UK HPR1000 design will be capable of reducing external dose rates SFAIRP.

4.4 Worker Dose Assessment

4.4.1 Assessment

156. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I reviewed and assessed the RP submissions in relation to occupational worker dose.

157. Dose limitation within a new nuclear facility is required within UK under the IRR17 (Ref. 10), whereby Regulation 12 (dose limitation) stipulates that "every employer must ensure that its employees and other persons within a class specified in schedule 3 are not exposed to ionising radiation to an extent that any dose limit specified in Part I of that schedule for such class of person is exceeded in any calendar year".

Table 2: IRR17 Schedule 3 Part 1- classes of persons to whom dose limits apply (Ref. 10).

	Dose Limits ($mSvy^{-1}$)		
	Employees and trainees of 18 years and above	**Trainees aged under 18 years**	**Other persons**
Effective dose	20	6	1
Lens of the eye	20	15	15
Skin (averaged over 1 cm^2)	500	150	50

158. The SAPs (Ref. 2) provide further information regarding dose targets for new nuclear facilities against regulatory expectations and RGP. SAP NT.1 provides numerical targets and limits which a nuclear facility should be assessed against. These targets include the BSL which is sometimes the legal limit, and the BSO. The BSO is a matter of policy and is set at a point where the need for any further scrutiny of the risks/hazards by ONR would likely not be necessary. Tables 3 and 4 provide the BSL and BSO targets for both employees on site and other workers on site as well as any group on site.

Table 3: Target 1 - Normal operation for any person on the site (Ref. 2).

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 (IRR17). Normal operational doses should also be reduced ALARP.*	

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Table 4: Target 2 - Normal operation for any group on the site (Ref. 2).

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	

159. TAG NS-TAST-GD-043 ‘The ‘Radiological Analysis for Normal Operation’ (Ref. 4) provides further information regarding the targets stipulated within the above tables specifically paragraph 5.15 - 5.22.

160. Both of the above targets are consistent with IAEA safety standards (Ref. 81). Specifically, to the following fundamental principles outlined in the document:

- 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.

161. For the assessment of worker dose, these aspects along with the NT1 and NT2 stipulated within the SAPs (Ref. 2), shall form the bases of my assessment on worker dose.

162. Within the PCSR Chapter 22 (Ref. 3), the RP has the following sub-claim:

“The risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP”

163. This sub-claim is part of the overarching claim which encompasses workers and members of the public whereby during normal operation the risk from ionising radiation is reduced in line with current guidance. Several arguments are put forward to support this sub-claim whereby worker dose is one of those arguments. The RP provide information on this argument as well as evidence to corroborate this claim through

PCSR Chapter 22 Section 22.9 (Ref. 3), whereby the RP provides an overview of the following:

- Methodology, calculation and ALARP assessment of the collective dose; and
- individual doses to workers on site.

164. When the I undertook the initial assessment of information provided by the RP, there were several shortfalls which were:

- The dose data for the generic UK HPR1000 design did not compare favourably with PWRs worldwide.
- There had been no demonstration that a systematic approach to identifying all potential improvements that may lead to a reduction in radiation dose had been implemented for the generic UK HPR1000 design.
- No evidence was provided regarding what impact the generic UK HPR1000 design maintenance arrangements and design provisions, (i.e. plant design features, frequency, intrusiveness of inspections, space provisions etc.), and the impact of other manual tasks and operations, have on the extent of the anticipated collective occupational radiation exposures (Ref. 19).

165. Due to the above reasons, I judged there to be a potential regulatory shortfall associated with worker dose evaluation and RO-UKHPR1000-0035 (Ref. 19) was raised.

4.4.1.1 RO-UKHPR1000-0035

166. To resolve the RO, the RP stated they would provide the following within their resolution plan (Ref. 82) :

- Review the approach for OPEX data selection, refine the starting point of collective dose evaluation and provide relevant information to ONR.
- Perform a gap analysis to assess the differences between Chinese PWRs and the UK regarding occupational exposure (such as on maintenance and operational tasks, etc.) and provide an explanation for how they are anticipated to influence (or not) occupational exposure for the generic UK HPR1000 design.
- Provide evidence by a suite of ALARP documents produced by SSC design areas and Radiological Protection area to demonstrate the application of the ALARP process for radiological protection to minimise occupational exposure and reduce it to ALARP.
- Establish linkages to SSC design area documents as evidence to substantiate the improved design features from the M 310 and CPR1000 plants which have been incorporated into the generic UK HPR1000 design and confirm that the influence from these improved design features have been evaluated and reflected in the collective dose; and
- Assess the impact on collective dose from the design modifications implemented in the design reference (DR) 2.2.

167. I assessed the following documents as part of my assessment of RO-UKHPR1000-0035 (Ref. 19):

- ‘Worker Dose Evaluation Topic Report’ (Ref. 79, Ref. 83)
- ‘Establishment of the Starting Point for the Collective Dose Evaluation of the UK HPR1000’ (Ref. 84)
- ‘Examination Maintenance Inspection Testing (EMIT) Consistency Analysis’ (Ref. 85)
- ‘EMIT Strategy Implementation Report’ (Ref. 86)

- ‘Evaluation of the Impacts on Collective Dose from the Design Improvements’ (Ref. 87)
- ‘ALARP Demonstration of PCSR Chapter 22’ (Ref. 16)
- ‘Source Term Reduction Analysis Report’ (Ref. 88)
- ‘Minimisation of the Occupational Exposure for RPV Head Assembly Lifting’ (Ref. 89)
- ‘Design Considerations to Minimise the Worker Dose for Valve Inspection and Maintenance’ (Ref. 90)
- ‘Design Considerations to Minimise the Worker Dose for Steam Generator Inspection and Maintenance’ (Ref. 91)
- ‘Layout Design Considerations to Minimise the Worker Dose’ (Ref. 92)
- ‘Optimisation in Occupational Exposure of Intermediate Level Waste (ILW) Spent Resin and Filter Cartridge Management’ (Ref. 93)
- ‘Process Risks/Hazards Analysis for In-Core Instrument Assembly (ICIA) Retrieval and Processing Operations’ (Ref. 94)
- ‘Process Risks/Hazards Analysis for ICIAs Packaging, Handling and Storage Operations’ (Ref. 95)

168. Following my initial assessment there were still issues identified hence the following RQs were raised (Ref. 20):

- RQ-UKHPR1000-0845
- RQ-UKHPR1000-1280
- RQ-UKHPR1000-1493
- RQ-UKHPR1000-1516
- RQ-UKHPR1000-1517
- RQ-UKHPR1000-1554
- RQ-UKHPR1000-1603

169. From the RP responses to the above RQs and discussions over the Step 4 of GDA period, the following outcomes were identified, and form part of the closure note for RO-UKHPR1000-0035 (Ref. 96).

170. The RP provided information that the preliminary worker collective dose for the generic UK HPR1000 design, is weighted towards early generation Chinese PWRs, and did not take into account the design improvements with the next generation of Chinese PWRs. From the RP review undertaken, which followed an appropriate methodology to establish the starting point for collective dose evaluation, the new collective worker dose for the generic UK HPR1000 design is comparable with similar sized PWRs (in terms of power output), around the world.

171. EMIT tasks make up most of the occupational exposure, compared to operational tasks. Hence the RP undertook a gap analysis in relation to EMIT task arrangements. From the RP review, there were no inconsistencies identified from a radiological protection perspective. However, I identified inconsistencies from a layout design perspective, which could indirectly affect radiological protection. I raised a matter whereby “Further evidence is required to demonstrate the inconsistencies identified from a layout design have minimal effect on collective worker dose”. This is discussed in more detail in section 4.4.1.2.

172. The RP provided the methodology of their ALARP process and how it is broken down into both “Holistic” and “Specific” ALARP assessment. The RP provide traceability to corroborate claims and arguments made, relating to design improvements implemented for the generic UK HPR1000 design.

173. Through documents I sampled, I judged the implementation of ALARP measures had been undertaken. This will be discussed further in Section 4.4.1.3.

174. Some aspects of the generic UKHPR1000 design are still in concept design. An example from my assessment includes engineering aspects (e.g. winding machine, shielding cover) involved in the removal of the ICIA. These are key SSCs, which the RP has placed claims which identify safety case requirements to demonstrate worker dose is reduced to ALARP. Considering the hazards involved in ICIA management, the significant contribution these operations make to collective worker doses, and following discussions with ONR's NLR assessor, I raised a matter whereby "Further evidence is required to demonstrate that the design of the key SSCs for waste ICIA retrieval operations, which the UK HPR1000 generic safety case identifies requirements for, will optimise the collective worker dose and reduce worker doses to ALARP." This will be discussed in more detail in Section 4.4.1.2.

175. The RP has provided a high-level overview of the key improvements for the collective dose for the generic UK HPR1000 design for DR1.0 as well as DR2.2. The quantitative effect this has on the collective dose (both positive and negative), is provided by the RP, whereby key safety case documents are referenced which provide further detail on the improvements.

176. The RP provided information on the additional modifications which have influenced radiological protection from DR2.1 to DR2.2. This led to the collective worker dose for DR2.2 of the generic UK HPR1000 design to be reduced by 11% compared with the DR1.0 value.

177. From the assessment of information provided by the RP, I was content in closing out RO-UKHPR1000-0035 with two matters (Ref. 96).

4.4.1.2 Matters from the Closure of RO-UKHPR1000-0035

178. From my assessment of RO-UKHPR1000-0035, two matters were raised (Ref. 96):

- Further evidence is required to demonstrate the inconsistencies identified from a layout design have minimal effect on collective worker dose.
- Further evidence is required to demonstrate that the design of the key SSCs for waste ICIA retrieval operations, which the UK HPR1000 generic safety case identifies requirements for, will optimise the collective worker dose and reduce worker doses to ALARP.

179. Regarding the first matter, within the response to RQ-UKHPR1000-1603 (Ref. 20), the RP has made a commitment to resolve this shortfall during site-specific stages. I consider this matter to be a minor shortfall.

180. Regarding the second matter, this is one of the more significant activities that has been assessed by ONR. Consequently, NLR raised RO-UKHPR1000-0037 (Ref. 19) to ensure ONR's regulatory expectations were understood and to seek improvements in the RP's safety case. Within the closure note for RO-UKHPR1000-0037 (Ref. 97) a similar matter from an NLR perspective has been raised. This is in relation to relevant risks being reduced to ALARP when the future licensee is managing ICIAs.

181. From a worker dose perspective, this is one of the work activities with a high risk of exposure, whereby further substantiation is required. This is most notable in that the RP has not demonstrated that it could adopt additional engineering controls to further restrict exposure rather than relying on administrative controls. Hence this is a shortfall in relation to IRR17 Regulation 9 (restriction of exposure). For this reason I judge this to be an AF.

AF-UKHPR1000-0101 – The licensee shall during detailed design, demonstrate that the design, operation and procedures for the removal of In Core Instrument Assemblies are optimised to reduce worker dose so far as is reasonably practicable.

4.4.1.3 ALARP Assessment of Worker Dose

182. The RP has followed a systematic approach for the calculation of the collective worker dose, as can be seen in figure 3.

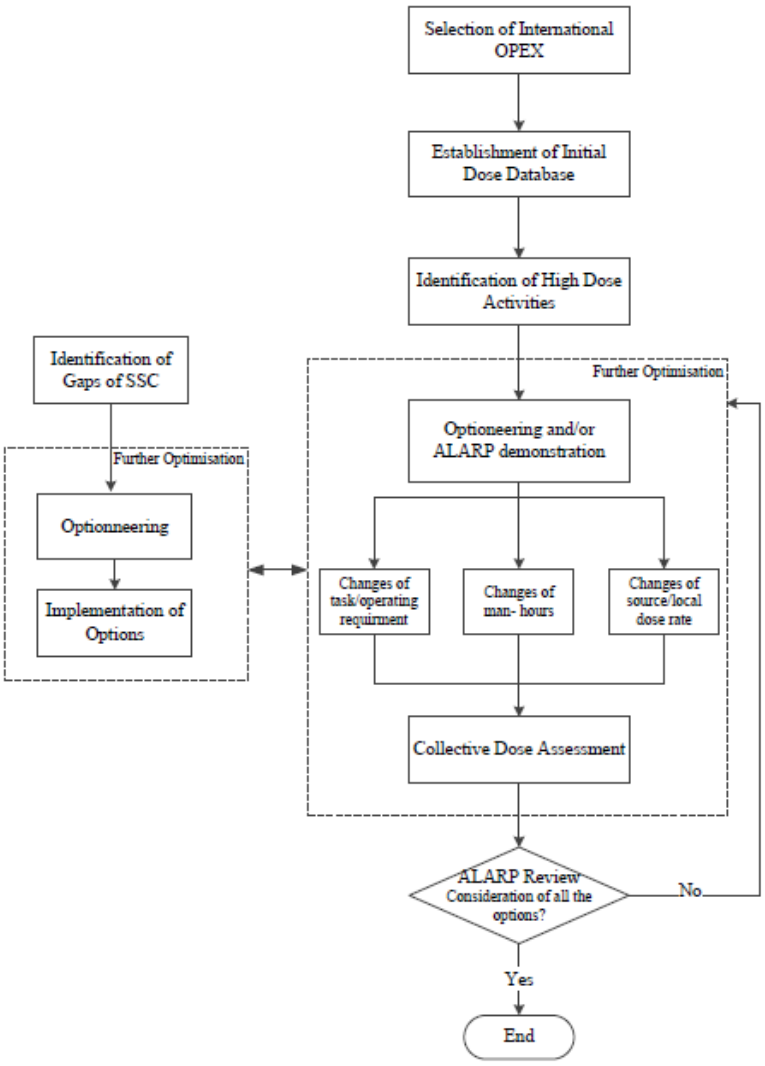

Figure 3: Logic diagram showing the approach undertaken to calculate the collective worker dose for the UK HPR1000 (Ref. 79).

183. The RP used appropriate OPEX information when calculating the initial collective worker dose for the generic UK HPR1000 design which covered previous evolutions of nuclear plants in China as well as PWRs around the world. However, the initial annual collective worker dose for the generic UK HPR1000 design was weighted towards early generation Chinese PWRs and did not take into account the design improvements with the next generation of Chinese PWRs. This approach generated the initial annual collective dose of 594 man.mSv/yr, However the RP revised the initial collective worker collective dose taking into account later design improvements which led to a collective dose of 370 man.mSv/year. The RP provided information which compared this collective worker dose against other operating PWRs around the world. (Ref. 84).

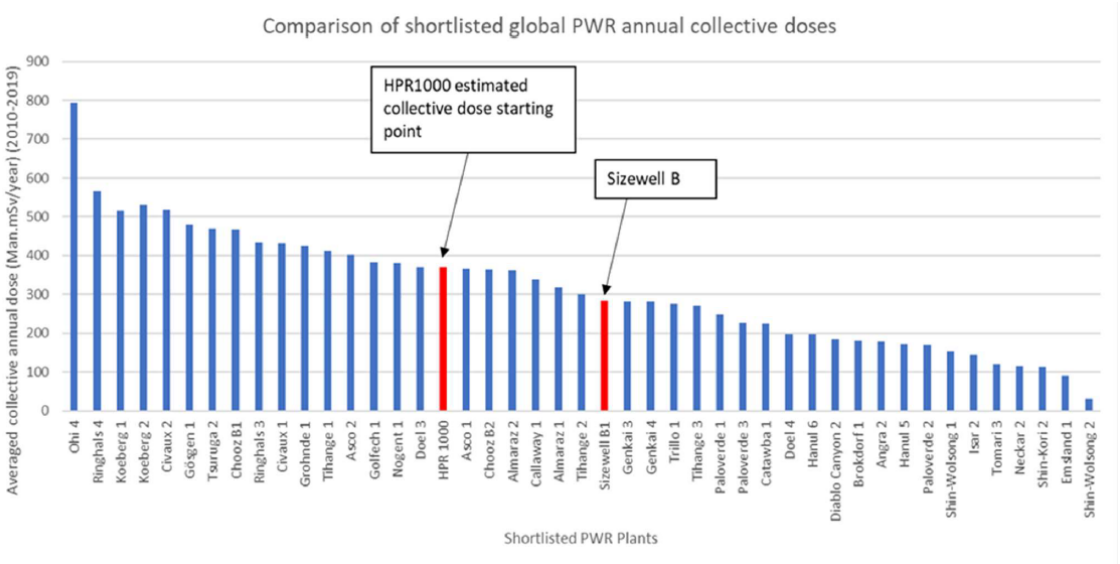

Figure 4: Graph comparing the collective worker dose for PWRs globally *(Ref. 84)*.

184. From figure 4, the initial worker collective dose for the generic UK HPR1000 design is comparable with other PWRs around the world. Although the generic UK HPR1000 design has a slightly higher worker collective dose than Sizewell B, the predicted collective worker dose for generic UK HPR1000 design is lower than the EPR™ (Ref. 98). This does not consider further worker dose optimisation which is discussed later on in this section.

185. The RP broke down the worker collective dose into various activities during normal operations and through OPEX were able to identify the six high dose activities for the generic UK HPR1000 design which were (Ref. 79):

- Works involving the valve inspection and maintenance
- On-site service
- Works involving the RPV
- Works involving the steam generator and reactor coolant pump
- In-service-inspection
- Waste processing

186. This comprises about 60% of the collective worker dose, the remaining 40% is dominated with large tasks that give rise to relatively low individual dose risks. These tasks will be discussed later in this section.

187. The next stage undertaken was to review and optimise the design via optioneering potential improvements to the design. Hence from DR1.0 to DR2.2 a number of design improvements were recognised which would impact the collective worker dose. In total 84 design improvements were recognised, 20 of which affect collective worker dose for the generic UK HPR1000 design. These were put into the following 13 categories (Ref. 87):

- Source term design improvements
- BQF
- Modification of spent fuel delivery process
- BQZ
- Modification of Solid waste treatment system
- Addition of concrete tank in the liquid waste treatment system
- Insulation material replacement in containment
- Reactor coolant pump design at GDA
- Valve number changes

- Improvements in welds
- Heating, Ventilation and Air Conditioning (HVAC) systems diversity modification;
- Diversity improvements
- Modification of steam generator blowdown system

188. The RP provided information (Ref. 87) on how the above design improvements affect the collective worker dose. From the assessment of these design improvements the following are to be noted.

189. Regarding source term design improvements, from a radiological protection perspective, the RP claim this is one of the major design improvements for the generic UK HPR1000 design and the biggest contributor to the reduction in collective worker dose (about 16%) (Ref. 79). However, from my assessment of the 'Source Term Reduction Analysis Report' (Ref. 88), as well as reviewing the RP's response to RQ-UKHPR1000-1493 (Ref. 20), I noted that further improvement beyond the 16% already stated is feasible. The information required will only become available once the generic UK HPR1000 design is operational. Hence, for this reason the RP has provided a conservative value on source term optimisation. The RP has stated it has taken a post Step 4 of GDA commitment to review international OPEX to further optimise their source term. I consider this matter to be a minor shortfall.

190. The addition of the BQF and the BQZ is a requirement for building a NPP within the UK. Although worker collective dose has been provided (based on UK OPEX information), further work will need to be undertaken to provide a more robust demonstration of worker dose optimisation. As these facilities are in conceptual design during Step 4 of GDA, detailed design will be undertaken as normal business during site-specific stage. I judge no further action needs to be undertaken during GDA.

191. From an NLR perspective, two ROs were raised regarding these facilities (RO-UKHPR1000-0040 and RO-UKHPR1000-0050 (Ref. 19)). From discussion with the NLR assessor one radiological protection recommendation was included during the closure of RO-UKHPR1000-0040 (Ref. 19), which was that the RP should demonstrate that the radiological risks from the layout of the BQZ are reduced to ALARP. After discussions with the NLR assessor, it was agreed that this will be taken forward during site-specific stages; hence no further action needs to be undertaken during Step 4 of GDA.

192. A further design improvement relating to radioactive waste management was the filter cartridge machine which was a significant design improvement compared to the CPR1000. The filters are now removed via the Spent Filter Cartridge Change Machine (SFCCM) which provides remote operation and radiation shielding for workers during filter replacements (Ref. 87), reducing the worker collective dose during this procedure.

193. The largest increase in collective worker doses (nearly 40 man.mSv per year) is attributed to the number of additional valves incorporated into the generic design , (partly due to compliance with HSG 253 (Ref. 99) (see Section 4.4.2 for further details)) which will require maintenance. RQ-UKHPR1000-0845 was raised to seek additional information on this matter. From the RP's response (Ref. 20) and supporting documentation (Ref. 87), the increase is stated to be due to the improvement of isolation and control of liquids, which in turn improves nuclear safety and plant control. The RP claims that it has, where possible, reduced the increase in occupational dose by taking the following mitigating steps (Ref. 87):

- Where possible replaced difficult to maintain valves with a simpler design;
- optimised valve layout;
- increased valves to help drained down areas where maintenance is required; and

- valves in high dose rate areas now have remote manual valves in lower dose rate areas.

194. The above does not consider the benefits of source term optimisation, which reduces the worker collective dose. The RP has provided information on how the occupational dose is affected by the addition of the valves and evidence that it is not feasible to further reduce the number of valves within the generic UK HPR1000 design. Consequently, no further action needs to be undertaken during GDA.

195. The design improvements were incorporated into the generic UK HPR1000 design and have an effect (as discussed above) on the high dose activities. For each of the activities further work was undertaken to reduce the worker collective dose using the ALARP methodology.

196. The RP used the Eliminate, Reduce, Isolate, Control and Personal Protective Equipment (ERICPPE) approach to reduce exposure to workers. The 'Worker Dose Topic Report' (Ref. 79) provides a high-level overview of the ALARP measures and further detailed information is provided within the reference documents for each of the high dose activities. Several of the measures undertaken are described below:

- To eliminate worker involvement the RP has used robotics where possible to undertake high dose work, an example would be to review the inspection welds on the SG (Ref. 91).
- Reduction of worker involvement by optimising the primary circuit temperature monitoring technique, this has led to 39 manual valves being removed along with relevant piping systems (Ref. 90).
- When installing new seal gaskets on the RPV closure head, temporary shielding is used to reduce worker dose (Ref. 89).

197. Where the RP considered that it was not reasonably practicable to adopt further engineering controls to restrict exposure, it has explained that further dose controls will be delivered by use of administrative controls and PPE, which will be the responsibility of the licensee. The RP has provided examples of these measures (Ref. 79) which are discussed in Section 4.6.

198. Within NS-TAST-GD-005 (Ref. 4) it states for nuclear new build that options should be identified for further improvements. The RP has provided evidence of the ALARP methodology used to reduce worker collective dose SFAIRP for the highest worker dose activities for the generic UK HPR1000 design. This amounts to roughly 60% of the total collective worker dose.

199. The RP has stated that dose minimisation for the remaining 40% of collective dose attributable to routine tasks can only be achieved once detailed design has been completed. Doses will then be optimised still further by application of operational strategies including guidance generated by the RP (Ref. 79) during Step 4 of GDA. This guidance includes safe system of work aspects which will be undertaken during the site-specific stage.

200. Although the evidence provided for the 40% of worker collective dose attributed to routine tasks may be minimal at present, it is proportionate for Step 4 of GDA. There are aspects related to detailed design which are not available at this stage, and doses will be further optimised by implementation of operational strategies. Consequently I accept the RPs argument that the remaining 40% of collective worker dose can't realistically be reduced at present within Step 4 of GDA. It is my judgement that further work needs to be undertaken with regards to dose assessment for routine tasks at the site-specific stage, and as such I have raised the following AF:

AF-UKHPR1000-0102 – The licensee shall during detailed design, complete the assessment of, and demonstrate that, collective worker dose for the UK HPR1000 design in totality, is reduced so far as is reasonably practicable. This should include the 40% of collective dose arising from routine activities which require detailed design and operational strategies to be addressed adequately.

201. In summary, due to the improvements the RP has undertaken with the generic UK HPR1000 design and implementing its ALARP methodology, the worker collective dose has been reduced from 370 man.mSv/year to 335.80 man.mSv/year during the GDA process which is a 9% reduction.

202. From my assessment the RP has demonstrated application of the ALARP methodology as stipulated within the safety case, in calculating the collective worker dose. In addition, the RP has provided examples of the ALARP assessment being undertaken for the high dose activities. As noted earlier, some aspects (most notably the radioactive waste management system as well as routine tasks) will require further ALARP demonstration, to reduce collective worker dose SFAIRP.

4.4.2 Review of Unmitigated Dose Assessment to Support HSG253 Compliance for the UK HPR1000

203. The following summarises an assessment of the queries and responses raised during a review of the RP's compliance with HSG253 with respect to radiological protection (Ref. 99). Health & Safety Executive guidance 'The Safe Isolation of Plant and Equipment' provides a framework for assessing and specifying adequate isolation and drainage of systems undergoing maintenance. The guidance provides a means of assessing the risk of harm to workers in the absence of isolation protection (unmitigated risk) from which a baseline isolation protection can be inferred. This guidance focusses on conventional risk such as high/low temperature, high pressure, chemotoxicity; however, it does not provide guidance on assessing unmitigated radiological risk.

204. My assessment entailed an initial review of the RP's 'Compliance with HSG253' report (Ref. 100) which summarises the RP's assessment of the design against HSG253 and responses to RQ-UKHPR1000-0389 (Ref. 20). This was followed by a subsequent review of the 'Unmitigated Radiological Dose Detailed Calculation' (HSG253 RPE Vent and Drain System [VDS] Sample) report (Ref. 101) and responses to a further RQ-UKHPR1000-0981 (Ref. 20) regarding the RP's methods for assessing unmitigated dose to support HSG253 compliance.

205. The scope of my assessment is confined to the review of the RP's assessment of radiological risk (i.e. unmitigated doses) which may be used to determine baseline isolation to support HSG253 compliance. This assessment takes into consideration the following regulations and guidance when assessing the RP's methods for determining unmitigated doses:

- SAPs (Ref. 2),
- IRR17 (Ref. 10),
- NS-TAST-GD-002 'Radiation Shielding' (Ref. 4), and
- Health and Safety Guidance for "The Safe Isolation of Plant and Equipment" (HSG253) (Ref. 99).

206. I considered submersion dose calculation methods to be suitable, however, the RP had incorrectly applied the geometric adjustment factor to submersion doses. The impact of this error was small as the submersion doses are generally much lower than the associated inhalation doses. The RP took actions to revise all calculations and updated documents accordingly (Ref. 102). I considered the revised submersion dose estimates to be reasonable.

207. **I**nhalation dose calculations employed suitable methods; the RP has used DCFs taken from 'Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors For Inhalation, Submersion and Ingestion' (Ref. 103) to estimate inhalation doses, as opposed to those currently recommended by ICRP. However, based on my assessment, it is my judgement that whilst DCFs from ICRP-119 (Ref. 104) should have been used, the impact on the unmitigated dose estimates used to underpin compliance with HSG253 will not impact the conclusions.

208. On the basis of my assessment of HSG253 I have identified a minor shortfall with respect to the RP not using the most appropriate DCFs for estimating inhalation doses. Although the RP provided evidence to state the difference between the RP guidance and current standards is minimal in this case, the most up to date standard should be used, which in this case is ICRP-119 (Ref. 104).

209. HSG253 generic assumptions were sufficiently conservative for the purposes of dose calculation (e.g. source terms, dimensions, worker position and exposure duration).

210. HSG253 direct dose calculations using MicroShield were both suitably conservative and fit for purpose.

4.4.3 Numerical Target 1

211. As stated already information relating to NT 1, is provided in Section 4.4.1.

212. The RP state (Ref. 79) the generic UK HPR1000 design will have an individual dose target whereby no worker will receive an annual effective dose exceeding 10mSv.

213. Although this is the dose target, the RP provided OPEX information of individual dose data from reactor units covering over 150 reactor years of operation. From this information the majority of workers were below 3 mSv/yr with only a small number above 10 mSv/yr, with none over the BSL of 20 mSv/yr (Ref. 79).

214. The RP provided further evidence whereby over the past couple of years the maximum individual dose to a worker recorded from CGN PWRs was below 10mSv/yr (Ref. 79). This again corroborates the claim the individual dose target for the UK HPR1000 of 10mSv is reasonable.

215. However, it should be noted that further potential improvements to the design at site-specific stages will be undertaken, which is expected to lower the maximum individual doses received to workers. With this in mind, and in addition to the annual effective dose target set by the RP, I judge the maximum individual dose to a worker for the generic UK HPR1000 safety case to be reasonable.

216. Regarding other employees on site, the RP has set an individual dose target of 1 mSv a year, this is below the BSL.

217. The RP has calculated the dose other employees on site will receive via identifying the main radiation source in the generic UK HPR1000 design and calculating exposure in un-designated areas. Hence the annual dose to other employees on site has been calculated as 7.28×10^{-3} mSv/year. This is below the BSO for normal operation for other employees on site, hence I am satisfied that the RP has demonstrated that it has met regulatory expectations with regard to the dose to other employees on site for the generic UK HPR1000 (Ref. 79).

4.4.4 Numerical Target 2

218. As stated already information relating to NT 2 is provided in Section 4.4.1.

219. From the assessment of the 'Worker Dose Topic Report' (Ref. 79) the RP provided evidence the group with the highest average exposure is for thermal insulation operations where the collective dose has been calculated with improvements in the generic UK HPR1000 design as 19.17 man.mSv/yr. This is then averaged between 12 workers (minimal amount of people required to complete this operation) which leads to an average dose of 1.6 mSv/yr. The RP explained that this type of work is similar to work undertaken in comparable reactors (e.g. Sizewell B).

220. The information provided by the RP is below the BSL, although above the BSO for NT2. I am satisfied that the RP has demonstrated that it has met regulatory expectations with regard to the average effective dose to a group of workers on site for the generic UK HPR1000.

4.4.5 Dose to the Eyes

221. Within IRR17 (Ref. 10) the equivalent dose limit for the eye was reduced from 150 mSv/yr to 20 mSv/yr. The RP has undertaken an OPEX review and undertaken theoretical calculations of typical working areas for the equivalent dose assessment for the lens of the eye.

222. It was noted the RP has minimal OPEX information relating to doses to the lens of the eye during normal operations. The information provided only covered several plants within China where research on the dose equivalent of the eye has been undertaken with no information on international OPEX.

223. The RP has been able to identify high dose risk activities to the lens of the eye using Chinese OPEX information, as well as using a modelling program to calculate the equivalent dose to the lens of the eye. From the modelling information, the RP has reviewed this against calculated actual dose accrued by workers to the eye and found the results were comparable. I judge this to be acceptable.

224. From the above calculations the RP has calculated a ratio factor between equivalent dose to the lens of the eye to equivalent dose to the body. From the RP calculations the equivalent dose to the lens of the eye will be below the legal limit of 20 mSv/yr as stated within IRR17 (Ref. 10). Although this is below the legal limit for dose equivalent to the lens of the eye, the RP has provided minimal information on the ALARP measures to be undertaken to reduce equivalent dose to the eye. Due to this shortfall as well as minimal international OPEX used, I judge this to be an AF:

AF-UKHPR1000-0103 – The licensee shall demonstrate that the doses received by a worker to the lens of the eye have been reduced so far as is reasonably practicable. Given recent legislative changes in dose limits to the lens of the eye, this should include addressing the shortfalls identified during Step 4 of GDA, including but not limited to, identification of reasonably practicable measures to reduce doses to the lens of the eye and the use of international operational experience.

4.4.6 Strengths

225. From the above assessment as outlined in Section 4.4, the following strengths have been identified relating to worker dose assessment:

- The RP has provided an appropriate methodology detailing the initial worker collective dose for the generic UK HPR1000 design.
- The RP has reviewed each of the design improvements and described how they affect the collective worker dose.
- The RP has undertaken an appropriate methodology to review the high dose activities to ensure that the worker collective dose is restricted SFAIRP.

- The collective worker dose for the generic UK HPR1000 design is comparable with other PWRs around the world.
- The OPEX information of the maximum individual dose to a worker is below the RP's target level.
- The dose to other employees on site during normal operation is below the BSO.
- HSG253 generic assumptions were sufficiently conservative for the purposes of dose calculation.

4.4.7 Outcomes

226. RO-UKHPR1000-0035 was closed with two matters.

227. The RP predicted collective worker dose for the UK HPR1000 is lower than the initial predictions at the start of Step 4 of GDA.

228. I have identified three minor shortfalls as discussed in the above sections.

229. I have identified three AFs which are:

- AF-UKHPR1000-0101 – The licensee shall during detailed design, demonstrate that the design, operation and procedures for the removal of In Core Instrument Assemblies are optimised to reduce worker dose so far as is reasonably practicable.
- AF-UKHPR1000-0102 – The licensee shall during detailed design, complete the assessment of, and demonstrate that, collective worker dose for the UK HPR1000 design in totality, is reduced so far as is reasonably practicable. This should include the 40% of collective dose arising from routine activities which require detailed design and operational strategies to be addressed adequately.
- AF-UKHPR1000-0103 – The licensee shall demonstrate that the doses received by a worker to the lens of the eye have been reduced so far as is reasonably practicable. Given recent legislative changes in dose limits to the lens of the eye, this should include addressing the shortfalls identified during Step 4 of GDA, including but not limited to, identification of reasonably practicable measures to reduce doses to the lens of the eye and the use of international operational experience.

230. The conservatisms employed within the RP's methodology for HSG253 (e.g. source terms, exposure duration, worker position) will offset the potential under-estimates arising from DCFs for inhalation. I therefore consider the RP's approach to calculating the radiological risk in support of HSG253 compliance to be reasonable.

231. Regarding the SAPs NT 1 and 2:

- NT 1 for maximum dose to an individual during normal operations is above the BSO though broadly acceptable.
- NT 1 for other employees on site during normal operation is below the BSO.
- NT 2 for a group of workers during normal operations is above the BSO though broadly acceptable.

232. The maximum eye dose to a worker during normal operations is below the legal limit but further work is required to demonstrate that the dose has been reduced SFAIRP.

4.4.8 Conclusion

233. Whilst I have identified three AFs, based on the outcome of my assessment of the worker dose, I have concluded:

- The RP has provided adequate demonstration that worker doses have been restricted SFAIRP for the generic UK HPR100 design.
- The RP predicted collective worker dose for the generic UK HPR1000 design is lower than the initial predictions at the start of Step 4 of GDA.
- The maximum worker dose during normal operations for the generic UK HPR1000 design is reasonable.
- The maximum dose to a group of workers during normal operation on site for the generic UK HPR1000 design is reasonable.
- Worker dose for other employees on site during normal operation for the generic UK HPR1000 design is reasonable.
- The RP has provided appropriate arguments and evidence for high dose activities to support the sub-claim made within the PCSR Chapter 22 (Ref. 3).
- The RPs approach to calculating the radiological risk in support of HSG253 compliance is adequate for GDA.

234. It is my opinion that the three AFs identified can be resolved by the licensee during site-specific stages.

4.5 Public Dose Assessment

4.5.1 Assessment

235. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I reviewed and assessed the public exposure from direct radiation shine from the generic UK HPR1000 design during normal operations. This forms part of the public dose assessment which also includes dose to the public from gaseous and liquid discharges from the site which is assessed by the Environment Agency.

236. The regulation of public radiation exposure during normal operation is shared between the Environment Agency and ONR, where IRR17 (Ref. 10) is enforced by ONR and 'The Environmental Permitting Regulations (England and Wales) 2016' (EPR16) (Ref. 105) is enforced by the Environment Agency. IRR17 (Ref. 10) requires dose constraints to restrict exposure to ionising radiation at the planning stage where it is appropriate to do so. The guidance to IRR17 (Ref. 10) 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. 2) in relation to NT 3 (see table 5) 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 Public Health England Centre for Radiation, Chemicals and Environmental Hazards (PHE-CRCE) recommended that the dose constraint for members of the public from new reactors should be 0.15 mSv per year (Ref. 106).

Table 5: Target 3 for normal operation – any person off the site (Ref. 2).

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 (IRR17).*	

237. Within the PCSR Chapter 22 (Ref. 3), the RP has the following sub-claim:

"The risk to members of the public from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP."

238. This sub-claim is part of the overarching claim which encompasses workers and members of the public. The RP provide an argument and evidence to corroborate this claim through PCSR Chapter 22 Section 22.10, whereby the RP provides an overview of the dose assessment for public from direct radiation, the information provided is a high-level overview of the following (Ref. 3):

- Identification of main radioactive source terms;
- dose assessment (input data, computer code); and
- demonstration of SFAIRP.

239. The RP provided further documentation which will be discussed later in this section.

240. It is an ONR expectation that the measured dose rates near a “representative operational reactor” are consistent with the measurable range of background radiation dose rates. This information is necessary to support the RP’s claim that doses to members of the public from direct radiation have been reduced SFAIRP. From the initial assessment of information provided by the RP, I judged there to be a potential regulatory shortfall as the RP was unable to provide actual dose rate measurements from a reactor site. As such, RO-UKHPR1000-0028 (Ref. 19) was raised, which is discussed in Section 4.5.1.1 below.

4.5.1.1 RO-UKHPR1000-0028

241. To resolve RO-UKHPR1000-0028, the RP stated they would provide the following information (Ref. 107):

- Identify a representative plant, or plants, to provide a meaningful comparison of the measured dose rates with the background radiation dose rates near the comparable reactor.
- Measure dose rates at a variety of distances whilst the plant is operating, in areas where background radiation levels are accuretly known.
- Provide analysis of any differences between measurement and calculation, taking into account any differences between the operational reactor being measured and the generic UK HPR1000 design, such as reactor power and radiation shielding design etc.
- Carry out a systematic determination of the background radiation dose rates using appropriate measurement devices stating what the errors are in any given data set and clearly stating what is included in background measurents and what has been subtracted. This may be by using data from prior to construction or using data obtained during outages. This data must be referenced.
- Provide a demonstration that the measured dose rates are within the measurable range of background radiation dose rates near the comparable reactor.
- Assess the impact of other buildings, such as radiactive waste and spent fuel stores on dose rates, drawing conclusions for the generic UK HPR1000 design.
- Include this information in the generic UK HPR1000 safety case.

242. The RP submitted the above information within the following documents:

- ‘Analysis of Environmental Dose Rate Measurements from the Representative Reactor’ (Ref. 108)
- ‘Public Dose Evaluation from Direct Radiation Topic Report’ (Ref. 109)

243. From the initial assessment, clarification was still required, hence RQs were raised (RQ-UKHPR1000-0838 and RQ-UKHPR1000-0850 (Ref. 20)).

244. From the RPs response to the above RQs and discussions over Step 4 of GDA, the following were raised within the assessment note to close RO-UKHPR1000-0028 (Ref. 110).

245. The RP provided a detailed history of background radiation dose rates within the vicinity of the “representative reactor” before and during operation, which covered a 16-year period. From the study there was no distinguishable difference between background and measured dose rates for the “representative reactor”.

246. The RP provided dose rates from 0.5 km from the “representative reactor” to over 7 km away, whereby this showed no distinguishable difference in readings when the “representative reactor” is in operation or outage. However, the RP could not provide dose rates from 0 to 0.5 km from the “representative reactor” as there is no requirement under applicable regulations in China to monitor the immediate surrounding of reactor units, or the production area. Information of this nature would be useful for calculating the exposure to a representative person (potentially a dog walker who walks along the nuclear site everyday), I consider this matter to be a minor shortfall.

247. The RP provided OPEX from the “representative reactor” against the generic UK HPR1000 design for several topic areas (i.e. source terms, radiation shielding configurations, additional facilities). From the evidence provided the document states the public dose from the generic UK HPR1000 design will be lower than the “representative reactor”.

248. The BQF and the BQZ provide about 95% of the direct radiation an adult would receive 100m from the site at an elevation of 1 m (6 µSv a year). However, these are currently in conceptual design and several assumptions have been made (i.e. the facilities are full of waste for its entire lifespan). Further work will be undertaken during site-specific stages to refine direct radiation dose for these buildings, which will be normal business. Hence a detailed ALARP assessment of these facilities are out of scope of this assessment.

249. I assessed and sampled the RP’s calculations for direct radiation, more specifically, their assumptions of radiation shielding factors as well as ratio of time spent indoors to outdoors. Even in the worst-case scenario (an adult stays outside for the full time at an elevation of 1 m and 100 m from the generic UK HPR1000 design), their calculated dose is still below the BSO for NT 3. On this basis I judge the information provided by the RP to be adequate.

250. From the assessment of information provided by the RP I was content in closing out RO-UKHPR1000-0028 (Ref. 110).

4.5.1.2 ALARP Assessment of Direct Dose

251. The RP followed their ALARP methodology that has been implemented for the generic UK HPR1000 design to demonstrate public dose has been reduced SFAIRP.

252. From a holistic perspective as stated in the previous section, the RP provided information on representative reactors regarding public dose exposure from the reactors up to a 7 km radius. From the report (Ref. 108) the evidence shows no distinguishable differences with background radiation and direct radiation from the site.

253. Regarding the design of the generic UK HPR1000 design, the RP demonstrated (Ref. 109) that the direct radiation shine to the public from the main nuclear island is a small proportion (less than 10%) of the total direct radiation dose to the public. Further design changes to reduce direct radiation dose in the main nuclear island would be in my judgement grossly disproportionate.

254. It is noted that the main source of direct radiation to the public is the BQF and BQZ facilities, which are in conceptual design at Step 4 of GDA. These facilities have not had the same level of scrutiny as the main nuclear island and as stated previously a detailed ALARP assessment is out of scope of Step 4 of GDA. From this aspect the BQF and the BQZ may require further design improvements so the direct radiation dose to the public is reduced SFAIRP. As it will be normal business at site-specific stages for these facilities to be further developed, I judge no further action needs to be undertaken during Step 4 of GDA.

255. On the basis of the information provided by the RP, whereby the RP stated the public dose from the generic UK HPR1000 design will be of the same or better than previous reactor designs (Ref. 109). In addition from my review of the above information provided against relevant guidance, I judge the RP has provided an adequate demonstration during Step 4 of GDA, to show main nuclear island public doses from direct radiation sources on site are reduced SFAIRP.

4.5.1.3 Numerical Target 3

256. From reviewing the direct radiation dose to an adult for the generic UK HPR1000 design, the RP values range from 6.3 µSv/year to 8.6 µSv/year depending on the elevation at 100 m away from the facility; this is below BSO NT 3.

257. Although the calculated dose from direct radiation shine from sources on site for the generic UK HPR1000 design is below the BSO for NT 3 (Ref. 2), this only forms part of the effective dose for any person off site. As stated, off-site doses resulting from discharges from the generic UK HPR1000 design also form part of the effective dose for any person off-site. Discharge doses are assessed by the Environment Agency during Step 4 of GDA and are included within the PCER Chapter 7 (Ref. 15).

258. When including the liquid and gaseous discharges as stated within PCER Chapter 7 (Ref. 15) to the direct radiation shine doses, this leads to the cumulative dose to a representative adult, child and infant of 22.8, 12.4 and 9.6 µSv/year respectively.

259. Looking at the above public dose rate values for a representative person, only the adult is above the BSO NT 3 whilst values for child and infant are below.

260. Although further work is to be undertaken during site-specific stages on reducing the dose to the public from direct radiation, for the purpose of GDA, I judge that the annual effective dose to the public from the generic UK HPR1000 design is reasonable.

4.5.2 Strengths

261. From the above assessment as outlined in Section 4.5, the following strengths have been identified relating to public dose assessment:

- The RP provided detailed history of background radiation dose rates for a representative reactor which range from 0.5 km to 7 km.
- OPEX from the representative reactor was used to compare against the generic UK HPR1000 design which provides evidence that the direct dose will be lower.
- Doses calculated for direct radiation dose are below NT 3.
- Dose rates calculated are broadly consistent with other reactor designs.
- An appropriate ALARP assessment has been undertaken to demonstrate the direct dose to the public from the nuclear island (excluding the BQF and BQZ) has been restricted SFAIRP.
- The combined off-site doses as well as the direct radiation dose from the generic UK HPR1000 design to a representative person are just above the BSO for an adult and below the BSO for child and infant.

- The RP has provided appropriate argument and evidence to corroborate the claim stipulate in the PCSR Chapter 22 (Ref. 3).

4.5.3 Outcomes

262. The assessment of RO-UKHPR1000-0028 is closed.

263. I have identified one minor shortfall as discussed in Section 4.5.1 above.

264. The off-site dose to a representative person from the UK HPR1000 is comparable to the BSO for NT 3.

4.5.4 Conclusion

265. Based on the outcome of my assessment of the public dose, I have concluded that the off-site dose to a representative person is acceptable and fulfils the sub-claim stipulated with the PCSR Chapter 22 (Ref. 3).

4.6 Radiation and Contamination Zoning

4.6.1 Assessment

266. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I assessed the detailed radiation and contamination zoning layout of the generic UK HPR1000 design designated areas, as well as the detailed proposals for designation of radiation areas to ensure consistency with IRR17 (Ref. 10).

267. Designation of areas within a new nuclear facility is a key design aspect and is required within the UK under the IRR17 (Ref. 10).

268. IRR17 (Ref. 10) Regulation 17 (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.

269. Other aspects of IRR17 (Ref. 10) are also applicable; Regulation 9 (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 Regulation 8 (radiation risk assessment) a radiological risk assessment must be completed before any work involving ionising radiation may be undertaken.

270. The SAPs (Ref. 2) provide further clarity to help guide my assessment for the proposed new nuclear facilities. SAP RP.3 stipulates that where appropriate, designated areas should be further divided, with associated controls to restrict exposure and prevent the spread of radioactive material. SAP RP.4 stipulates that effective means for protecting persons entering and working in contaminated areas should be provided, and SAP RP.7 stipulates a hierarchy of control measures be applied to optimise protection.

271. Further requirements are stated within NS-TAST-GD-038 (Ref. 4) paragraph 5.7. The zone category should be an indication of the required degree of engineered and managerial controls and should increase e.g. Contamination zone 1 (C1), Contamination zone 2 (C2), Contamination Zone 3 (C3) and Radiation zone (R1), Radiation zone 2 (R2), Radiation zone (R3), etc., dependent on the increase in radiological and contamination risk. In paragraph 5.8 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.

272. Within the PCSR Chapter 22 (Ref. 3), the RP has the following sub-claim:

"The risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP"

273. This sub-claim is part of the overarching claim which encompasses workers and members of the public whereby during normal operation the risk to ionising radiation is restricted in line with current guidance. Several arguments are put forward to support this sub-claim whereby the radiation and contamination zoning arrangements is one of those arguments. The RP provide information on this argument as well as evidence to corroborate this claim through PCSR Chapter 22 Section 22.7 (Ref. 3), whereby the RP provides an overview relating to radiation and contamination zoning of the following:

- General description of undesignated / designated areas
- Radiation zoning and classification principles
- Contamination zoning principles
- Access and egress control
- Radiation classification

274. Further information is provided in the following documents which were assessed:

- 'Radiation and Contamination Zoning Topic Report' (Ref. 111)
- 'Optioneering Report of Contamination and Access Control Method' (Ref. 112)
- 'Lists of Rooms with Iodine and Aerosol Risks' (Ref. 113)
- Sample of radiation classification report
- Sample of radiation zones plan view

275. From the initial assessment of the above documents further information was required, hence the following RQs were raised (Ref. 20):

- RQ-UKHPR1000-0846
- RQ-UKHPR1000-0848
- RQ-UKHPR1000-1054
- RQ-UKHPR1000-1611
- RQ-UKHPR1000-1612
- RQ-UKHPR1000-1645

4.6.1.1 Radiation Zoning

276. The RP has split up the radiation zoning for the generic UK HPR1000 design into supervised as well as controlled area, however the controlled area is further split into four sub radiation zones.

Table 6 : Designation of areas for radiation in the generic UK HPR1000 design (Ref. 111).

Zoning		Area Dose Rate Limit (mSv/h)	Habitability
Undesignated area		≤0.0005	Indefinite
Supervised area (White)		≤0.0025	Quarterly working time less than 500 hours
Controlled area	Conventional working area (Green)	≤0.01	Weekly working time less than 40 hours
	Intermittent working area (Yellow)	≤1	Weekly working time less than 4 hours
	High radiation area (Orange)	≤100	Limited access
	Extremely high radiation area (Red)	>100	No entry

277. As can be seen from table 6, as the radiation level increases so does the restrictions and necessary mitigation measures to restrict exposure, to the point that areas classified as red zone are no entry in normal circumstances. For the purpose of compartment radiation classification, the zones are subdivided further to optimise radiation design (Ref. 111).

278. The RP has created a series of zoning principles to help specify the current zoning level for compartments and help reduce occupational exposure. The principles are (Ref. 111):

- Appropriate radiation shielding design is provided to consider potential radiological dose rates and anticipated access requirements for the area.
- Suitable and sufficient monitoring arrangements are provided to adequately monitor the radiation in these areas and this is kept under review.
- Appropriate boundaries and signage is provided.
- Specific design requirements of the layout of controlled area (workers will not need to go through a higher radiation / contamination zone to reach a lower radiation / contamination zone or adequate dimensions in the layout).

279. From reviewing the principles and the radiation zoning methodology, I judge this to be adequate and in line with current guidance as stipulated within IRR17 (Ref. 10) and SAP RP.3 (Ref. 2).

4.6.1.2 Contamination Zoning

280. The RP has taken a similar approach whereby contamination zoning like radiation zoning has been broken down into subcategories which are:

- C0 – Area, rooms free of loose and airborne contamination.
- C1 – Areas and rooms which are substantially clean and surface contamination is strictly controlled and periodically inspected.
- C2 – Areas and rooms where the risk of surface contamination is high.
- C3 – Areas and rooms with an airborne contamination hazard.

281. For each of the above contamination designated areas there are specific potential contamination limits which apply for different types of radiation, this is described in table 7.

Table 7: Designation of areas for contamination in the generic UK HPR1000 design (Ref. 114)

Area Designation	Potential Contamination Level‡	Typical area
C0	≤0.4 Bq/cm^2 (α/β/γ)	Supervised area
C1	≤0.4 Bq/cm^2 (α)	Most areas which are controlled areas
	≤4 Bq/cm^2 (β/γ)	
C2 (Surface activity)	>0.4 Bq/cm^2 (α)	Contaminated sumps and pools
	>4 Bq/cm^2 (β/γ)	
C3 (airborne activity, iodine and / or aerosol risk)	>0.01 Bq/m^3 (α)	Some RCV valve rooms and pipe rooms
	>10 Bq/m^3 (β/γ)	

282. As stated above C1 areas are classed as clean areas and are periodically monitored to keep contamination levels below the limits of C2. If the levels of contamination are above C1 limits and it is not possible to decontaminate immediately, then a temporary C2 area will be designated.

283. For C2 rooms the contamination will be controlled through local rules, plant operational controls and have physical barriers to prevent spread of contamination. Typical C2 rooms include the reactor pool, loading pit. During outage areas that will require maintenance could have the potential to be redesignated as temporary C2 areas.

284. Finally, C3 areas will be designated during the operational stage. These will be identified as areas with a potential for airborne leakage or during tasks such as equipment maintenance. Any potential iodine or aerosol contamination risks have been identified through the document ‘List of rooms with iodine and aerosol risks’ (Ref. 113). The rooms specified in this document will not necessary require C3 classification, however they have been designed to incorporate HVAC systems to mitigate airborne risk. If other areas are identified as C3 when UK HPR1000 is operational, temporary controlled areas with ventilation can be designated.

285. From a high-level design perspective, the following design requirements are required to help reduce occupational exposure (Ref. 111):

- Equipment and pipes containing radioactivity should be designed and manufactured to prevent leakage.
- Equipment vents and drains from radioactive systems should be fed directly to the collection and treatment systems.
- Epoxy paints and suitable smooth surface coatings to concrete floors and walls used to help decontaminated areas.
- Adequate monitoring measures undertaken to ensure levels of contamination are kept under control.

‡ It should be noted that the contamination level for ^{3}H is 10 times the corresponding beta level within the contamination zone as stated in the table, this is common practice within Chinese regulation (Ref. 93).

- Change rooms made available at access of controlled areas following appropriate control measures for change rooms (e.g. barrier, enough capacity to facilitate all possible needs).
- Contamination barriers at access to C2 zones.
- Adequate and sufficient ventilation systems to ensure airflow moves from lower to higher contamination zone, whereby air change frequency for C1, C2 and C3 is once per hour, twice per hour and four times an hour respectively.

286. From reviewing the principles and the contamination zoning methodology, I judge this to be adequate and in line with current practice as stipulated within IRR17 (Ref. 10) and RP.3 / RP.4 in SAPs (Ref. 2).

4.6.1.3 Radiation and Contamination Zone Maps

287. The RP has provided radiation zone maps for the following buildings:

- BRX – Normal operation
- BRX – shutdown
- BNX – Normal operation
- BSA/BSB/BSC – Normal operation
- BFX – Normal operation
- Radioactive Waste Treatment Building (BWX) – Normal operation
- Personnel Access Building.

288. It should be noted that apart from the BRX, other buildings unless stated otherwise, have the same zoning for normal operation (this includes during power as well as during shutdown). In addition some of the buildings have no radiation zoning maps provided. The RP has stated that these buildings are still in conceptual design (for example the BQZ), these will be completed during site-specific stages as normal business.

289. The radiation classification reports provide information on the zoning classifications for each compartment. Two sets of classification are used when considering the radiation classification, these are (Ref. 115):

- Typical class (calculated based on the realistic source term).
- Design class (calculated based on the design source term).

290. The RP further state the reasoning for two classifications is to demonstrate ALARP decisions as the typical class is based on realistic dose assessment, whilst the design class is used for biological shielding and room zoning. The radiation zone maps are based on the design source term (Ref. 115).

291. I have sampled the radiation zone maps as well as the radiation classification reports and have raised RQs where required (Ref. 20). Overall in my judgement the radiation zone maps follow the design principles as stated within Section 4.6.1.1. of my assessment.

292. It is noted, that from the assessment of RO-UKHPR1000-0060 (Ref. 19), the radiation zone maps have not been updated to consider changes in relation to radiation shielding. I consider this matter to be a minor shortfall that can be resolved during normal business at site-specific stages.

293. It is also noted that within the radiation zone maps the HVAC shafts and cable shafts which are accessible to workers have no radiation zoning. From the RQ response to RQ-UKHPR1000-1054 (Ref. 20), the RP stated it is not Chinese practice to designate HVAC and cable shaft areas, as these will not be accessed on a regular basis. I

consider this to be a minor shortfall as it is possible for a worker to receive a dose within the HVAC shafts and cable shafts if entry is required.

294. With regards to contamination zone maps the RP has stated none will be produced at Step 4 of GDA. The RP state that options will be up to the licensee on the choice of protective clothing requirements for entry into C1, C2 and C3 areas respectively. This could impact on the future design and layout of changerooms within the generic UK HPR1000 design. The RP has identified aspects of the design which would be permanent C2 areas (for example sumps and pools), though it has not identified areas which could potentially be C3. Although a contamination zone map would provide evidence to corroborate the high-level design principles as stipulated within Section 4.6.1.2 of my Step 4 of GDA Radiological Protection assessment, as this work will be completed at site-specific stages, I have considered this to be a minor shortfall.

295. From assessing the radiation zoning maps, I judge this to be adequate and in line with current practice as stipulated within IRR17 (Ref. 10) and RP.3, RP.4 SAPs (Ref. 2).

4.6.1.4 Access and Egress Control

296. The RP provided information on their approach to access and egress control for the generic UK HPR1000 design. The main principle is to mitigate the spread of contamination in controlled areas and protect workers from being exposed.

297. There are several access and egress controls in place for the following:

- Radiation controlled areas
- Contaminated areas
- Items
- High exposure risks

298. For radiation controlled areas before entering an area, a worker must undergo specific training as well as having a system of work in place (which will include a radiological risk assessment in line with IRR17 guidance). These aspects will be undertaken when the site is operational. When accessing the controlled area the worker will carry appropriate dosimetry (Electronic Personal Dosimeter (EPD) / passive dosimeter) and have the authorisation to enter the controlled area (through a turnstile). When in the radiation controlled area the worker will wear the necessary PPE (Ref. 111).

299. When exiting the radiation controlled area, radioactive waste material must be appropriately disposed of. Tools, equipment and personnel that entered the controlled area, must be monitored when leaving, to minimise the spread of contamination. If contamination is detected on personnel they are taken to the decontamination room for decontamination (Ref. 111).

300. For access to contamination areas, it is a similar procedure as when entering a radiation controlled area. When accessing the contamination areas the worker must have the correct dosimetry. Requirements for PPE are dependent on the contamination zoning:

- C1 is normal clothing
- C2 wear overshoes, gloves, coveralls
- C3 is further PPE as well as wearing appropriate Respiratory Protective Equipment (RPE)

301. The selection of appropriate PPE will be a decision for the licensee, although the RP has provided appropriate examples of PPE to be used (Ref. 111). I judge this to be adequate.

302. When entering controlled contamination areas workers will have to cross a boot barrier when going from one contamination zone to a higher contamination zone. Whereby if going from C1 to C2 the following items are required (Ref. 111):

- A physical barrier with clear signage
- Plastic cloth on C2 side
- Storage for PPE
- Waste container
- Container for re-usable contaminated clothing
- Inspection equipment
- Tacky mat on C1 zone to reduce spread of contamination

303. When entering airborne contamination rooms (C3) additional measures are required which are (Ref. 111):

- Containment is established with a well-sealed curtain to ensure negative pressure
- Ventilation to exhaust air from C3 zone
- Additional PPE equipment

304. When exiting a contamination area it is a similar procedure as exiting a radiation controlled area. Depending on the type of contamination zone the worker has entered (C1, C2 or C3), there may be requirements for additional changes and monitoring before exiting the contamination area (Ref. 111).

305. Small items (such as pens, keys and notebooks) that are carried into the controlled contamination areas by workers are monitored upon exiting to ensure that they are below specified limits (Ref. 111).

306. Large items are transported through a special passageway which has boundary gates, which are closed during normal operations. The large items are monitored by specially trained workers, to demonstrate that surface contamination levels are acceptable and below defined limits (Ref. 111).

307. For high exposure areas (orange and red zones) the RP has provided the following control measures (Ref. 111).

- From an engineered perspective all orange and red zones are locked with key management.
- To access an orange or red zone, workers have to first go through a green and yellow zones which have their own engineered controls (e.g. installed monitoring system).
- Have a permit to work system which controls the work, and the number of people present during the work:
    - Barriers and signs warning workers they are entering a high radiation area.
    - Where there is a transient high dose rate interlocks are considered and implemented as appropriate to prevent workers entering when the dose rate is high.
    - Temporary or permanent surface contamination control zone is established prior to entry into high radiation area.
    - Appropriate PPE must be worn when entering these areas.

308. From reviewing the access and egress controls stipulated by the RP for the generic UK HPR1000 design, I judge this to be adequate and in line with current practice as stipulated within IRR17 (Ref. 10) and SAPs (Ref. 2).

4.6.1.5 ALARP Assessment of Radiation and Contamination Zoning

309. The RP has undertaken an optioneering review and assessment of radiation and contamination zoning as well as access and egress control. The approach undertaken by the RP is provided in figure 5.

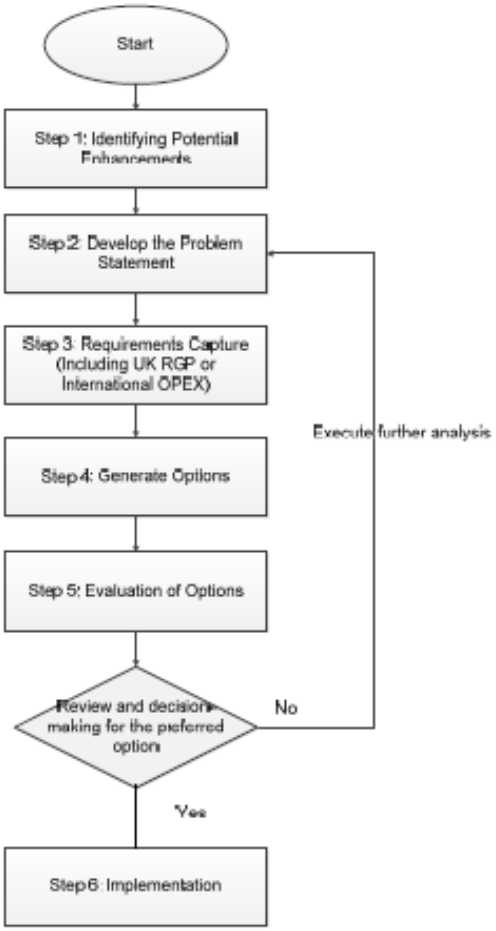

Figure 5: Optioneering process undertaken by the RP in relation to radiation and contamination zoning (Ref. 112).

310. The RP included the following assumptions when undertaking the optioneering process:

- The current regulatory legislation is assumed unchanged during the optioneering assessment.
- The resource to operate the generic UK HPR1000 design is the same for both full change or ‘entry in blue’.
- The assessment criteria is underpinned by a combination of CGN, EDF and GNSL OPEX.

311. The reason for this optioneering was due to a difference between Chinese and UK practice in relation to zoning strategy as well as the approach to access and egress control.

312. The RP reviewed current legislation such as IRR17 as well as SAPs, TAGs and industry code of practice such as the ‘Changeroom design, operation and maintenance - a nuclear industry code of practice’ (Ref. 116).

313. Although the main focus of this document is the review of access and egress into a nuclear facility, the RP has undertaken a review of radiation and contamination zoning within the UK nuclear industry and provided information on their findings.

314. The RP compared the access and egress methods as well as contamination control approaches used within the UK nuclear industry for several licensees in addition to the approach in China. It noted two approaches used, full change and 'entry in blue'.

315. Both approaches are used within the UK nuclear industry whilst in China only full change is used.

316. There are slight differences in contamination controls and monitoring between the two approaches. The RP undertook an assessment of both approaches against a particular set of criteria which covered a number of areas such as safety, environment and costs.

317. Each of the criteria chosen, had a specific weighting factor based on the significance of the assessment criteria ranging from 1 to 4. From the optioneering workshop the RP concluded that the full change approach scored higher than entry in blue, though the differences in score is low.

318. It should be noted that the licensee may consider going for 'entry in blue' instead of full change which means changes to the current design to accommodate this. The RP has stated that within the current design there is adequate space for entry in blue to be undertaken. However, the RP noted that further work will have to be undertaken to confirm that this is feasible. As this would be undertaken during site-specific stages, I judge no further action needs to be undertaken during Step 4 of GDA.

319. From my assessment of documents, the RP has undertaken a methodological approach in optioneering the access and egress control methods currently used within the UK as well as China. Although the approach chosen is different to current operational reactors within the UK, the RP has stated that it can be changed in the future if required.

320. Regarding the radiation and contamination zoning levels stipulated for the generic UK HPR1000, as stated already the RP reviewed current radiation and contamination zoning approaches undertaken within the UK, as well as other reactors that have undergone GDA. It was noted the approach undertaken by the RP is similar to the UK EPR™ regarding radiological and contamination zoning criteria.

321. The RP has provided specific information on the ALARP assessment for the generic UK HPR1000 design in relation to radiation and contamination zoning within the radiation shielding design, which was covered previously in section 4.3 of my Step 4 of GDA Radiological Protection assessment.

322. It is noted that radiation and contamination zoning is one of the approaches undertaken to 'control' radiation and contamination under ERICPPE. Throughout the assessment of the generic UK HPR1000 design the RP has provided evidence relating to control of radiation and contamination through the zoning approach. However, in most cases the control aspects will be for the licensee to implement. I judge this to be adequate for GDA.

323. From discussions with the NLR assessor, it is noted that some of the equipment to be operated within the generic UK HPR1000 design requires further work to improve containment of contamination. An example within RQ-UKHPR1000-1560 and RQ-UKHPR1000-1772 (Ref. 20), is the SFCCM, whereby there is no information on how containment is maintained for the machine when the filter housing is opened. There are also additional questions on other waste systems for the generic UK HPR1000

design regarding contamination control. The NLR assessor has raised the following AFs:

AF-UKHPR1000-0181 – The licensee shall provide adequate evidence that the detailed design of the Spent Filter Cartridge Changing Machine (SFCCM), relevant to containment and contamination controls, reduces the risks to ALARP, for both normal operations and in fault or accident conditions and that the SFCCM SCCs are adequately categorised and classified.

AF-UKHPR1000-0182 – The licensee shall provide adequate evidence that the detailed design of the Spent Filter Replacement and Transfer Device (SFRTD), relevant to containment and contamination controls, reduce the risks to ALARP, for both normal operations and in fault or accident conditions, and that the SFRTD SCCs are adequately categorised and classified.

324. As can be seen from the above AFs containment and contamination controls are required to reduce risk SFAIRP, for both normal operations and in fault or accident conditions. As this has been identified by the NLR assessor, I judge no further action needs to be undertaken from a Radiological Protection perspective.

325. From the information provided and assessed it is noted the RP has produced appropriate arguments to corroborate their sub-claim as stipulated within Section 4.6.1. of my Step 4 of GDA Radiological Protection assessment. I judge the methodology for radiation and contamination zoning for the generic UK HPR1000 design to be appropriate and consider that the radiation and contamination zoning for the generic UK HPR1000 design overall is adequate and follows the design principles.

4.6.2 Strengths

326. From the above assessment on radiation and contamination zoning as outlined in Section 4.6 of my assessment, the following strengths from the RP safety case for the generic UK HPR1000 design are:

- The RP has provided appropriate information to demonstrate compliance with current standards and guidance in relation to radiation and contamination zoning.
- The radiation and contamination zoning methodology is similar to other nuclear sites within the UK.
- The radiological zoning maps for the generic UK HPR1000 design are reasonable.
- Access and egress control methods for the generic UK HPR1000 design are reasonable.
- The RP has undertaken a review of current UK OPEX in relation to radiological and contamination zoning as well as access and egress controls.

4.6.3 Outcomes

327. I have identified three minor shortfalls as discussed in the above section.

328. The radiation and contamination zoning arrangements for the generic UK HPR1000 design are reasonable.

4.6.4 Conclusion

329. Based on the outcome of my assessment of the radiation and contamination zoning, I have concluded:

- The RP methodology for radiation and contamination zoning has demonstrated compliance with current standards and guidance.
- The RPs approach to access and egress control are reasonable and demonstrated compliance with the current standards and guidance.
- The RP has provided appropriate arguments and evidence for radiation and contamination zoning to support the sub-claim made within the PCSR Chapter 22 (Ref. 3).

4.7 Radiation and Contamination Monitoring

4.7.1 Assessment

330. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I assessed the latest revision of the 'Radiation and Contamination Monitoring Topic Report' (Ref. 117) along with the suite of documents making up the Plant Radiation Monitoring (KRT[PMR]) System Design Manual.

331. The main regulation which is applicable to radiation and contamination monitoring against which I assessed PCSR Chapter 22 (Ref. 3) is IRR17 (Ref. 10).

332. NS-TAST-GD-038 (Ref. 4), paragraph 5.5 and 5.6 provides information on the necessary dosimetry equipment to measure radiation doses that should be available to workers. The TAG also refers to SAPs (Ref. 2) paragraph 592 and 593 on the types of monitoring instruments and alarms required on a site.

333. Within the PCSR Chapter 22 (Ref. 3), the RP has the following sub-claim:

"The risk to workers from the potential harmful effects of ionising radiation during normal operation complies with UK legal requirements and is ALARP."

334. This sub-claim is part of the overarching claim which encompasses workers and members of the public, whereby during normal operation the risk from ionising radiation is restricted in line with current guidance. Several arguments are put forward to this sub-claim whereby monitoring arrangements is one of those arguments. The RP provide information on this argument as well as evidence to corroborate this claim through PCSR Chapter 22 Section 22.8 (Ref. 3). The RP provides an overview of the following:

- Functions
- Different types of monitoring at UK HPR1000
- ALARP demonstration

335. Further information is provided in the following documents which I assessed:

- 'Radiation and Contamination Monitoring Topic Report' (Ref. 117)
- 'KRT Area Radiation Monitoring Functional Report' (Ref. 118)
- 'KRT Post Accident radiation Monitoring Functional Report' (Ref. 119)
- 'KRT Plant Radiation Monitoring System Design Manual Chapter 6' (Ref. 120)

336. From the initial assessment of the above documents further information was requested from the RP in the following RQs (Ref. 20):

- RQ-UKHPR1000-0851
- RQ-UKHPR1000-1609
- RQ-UKHPR1000-1610
- RQ-UKHPR1000-1660
- RQ-UKHPR1000-1703

4.7.1.1 Monitoring Methods and Functions

337. For the generic UK HPR1000 design there are three methods for radiation and contamination monitoring (Ref. 117):

- Continuous monitoring;
- sampling analysis; and
- portable or mobile monitoring.

338. These methods of monitoring are undertaken through different types of monitoring processes, whereby they fulfil one or more of the following functions (Ref. 117):

- Identify abnormal changes of activity.
- Continuous monitoring of activity concentration of gas and liquid effluent.
- Continuous monitoring of activity level of gas and liquid effluent.
- Automation of some alarms when the activity level exceeds a certain threshold.
- Measuring dose rate received by workers and recording the accumulated dose.
- Monitoring the radioactive contamination level of workers and small item's when exiting a controlled area.

339. Table 8 provides an overview of the different radiation and contamination processes in operation within the generic UK HPR1000 design, as well as the function alarm settings and areas they cover.

Table 8: Radiation and contamination monitoring methods and functions for the generic UK HPR1000 design (Ref. 117).

Monitoring Method	Function	Alarm Settings	Areas monitored / sampled
Process Activity	Continuously monitor the dose rate or activity concentration of process fluid.	Set at specific thresholds to either inform the operator of an abnormal condition or to automatically take the necessary action to isolate.	■ Steam Generator Leakage Monitoring ■ Fuel Cladding Monitoring ■ Integrity Monitoring of Primary Pressure Boundary ■ Gaseous Waste Treatment system monitoring ■ Nuclear island Vent and Drain System Relay Tank and Relay Sump Monitoring ■ RCV[CVCS] Coolant Filter Monitoring
Effluent Activity	Monitor the gaseous and liquid effluent systems	There are a number of alarms. One alerts the operators to an increase whilst another informs the MCR and an isolation valve is activated automatically, when the threshold is exceeded or a failure detected.	■ Gaseous waste treatment system ■ HVAC ■ Condensate vacuum system ■ Nuclear Island Liquid Waste Discharge System ■ Conventional Island Liquid Waste Discharge System

Monitoring Method	Function	Alarm Settings	Areas monitored / sampled
Area Dose Rate	To warn operators of a rise in dose rate within work areas either through installed or portable equipment.	There are a number of alarms. One alerts operators whilst another activates if the threshold has been breached. Visual / audio warning will occur to warn workers to take the necessary action to mitigate the dose as well as the information being sent to the MCR. In some cases, if a monitoring threshold is breached this will trigger automation of ventilation system to mitigate radioactive substance being released to the environment.	■ Reactor pool ■ Refuelling Machine ■ Spent fuel pool ■ Spent fuel pool crane ■ Fuel handling and storage system transfer room ■ Filter replacement room ■ Hot laboratory ■ Sample analysis room ■ Hot mechanical workshop ■ Service corridors for BSA, BSB and BSC; ■ Near personnel air lock in safeguard building ■ Cement curing room within BWX ■ Waste drum transport and grouting room in BWX ■ Stores for drums of BWX.
Area Airborne	The activity concentration of noble gases, iodine, aerosols in work areas and ventilation system during normal operation.	An alarm activates if it detects radionuclides above a certain threshold providing audio / visual alarms to warn workers.	■ Activity concertation of noble gases, iodine and aerosol in the air of the BRX ■ Sampling of noble gases, iodine, aerosols and ^{3}H in air of BRX ■ Activity concentration monitoring of noble gases in exhaust air in Containment Sweeping and Blowdown Ventilation System (EBA[CSBVS) large flow ■ Activity concentration monitoring of noble gases in exhaust air in EBA low flow ■ Sampling of aerosol and iodine in exhaust air of the EBA low flow ■ Activity concentration monitoring of noble gases in exhaust air of nuclear island ventilation systems ■ Activity concentration monitoring of aerosols in exhaust air of the hot laboratory and hot mechanical workshop ■ Gamma dose rate monitoring of the MCR intake air ■ Activity concentration monitoring of aerosol and iodine in the exhaust air of the waste

Monitoring Method	Function	Alarm Settings	Areas monitored / sampled
			treatment building ventilation system
Accident and Post-Accident	Provide operators with information on measures to undertake to minimise exposure to workers and the public during an accident	There are some alarms which if activated above a certain threshold will automatically interlock areas to protect workers.	■ Gamma dose rate monitoring in BRX ■ Radioactivity monitoring of containment annulus exhaust air ■ Radioactive concentration monitoring of exhaust air released from containment ■ Radioactivity monitoring of BSA, BSB and BSC exhaust air; ■ Radioactivity monitoring of exhaust air in main stack ■ Radioactive concentration monitoring of noble gases in Containment Sweeping and Blowdown Ventilation System low flow exhaust air ■ Area dose rate monitoring in BFX ■ Steam generator leak and radioactive concentration of noble gases monitoring ■ Dose rate monitoring of MCR intake air.
Personal dose equivalent	Evaluate the workers effective dose as well as equivalent dose to irradiated organs	If above the dose threshold will go into alarm and the worker follows the emergency procedure.	To measure external dose use: ■ Passive Dosimeter ■ EPD. To measure internal dose: ■ In vivo (direct measurement of radionuclide in the whole body) ■ In vitro (analysis of excreta and other biological samples)
Surface Contamination	Prevent the spread of radioactive contamination from controlled areas.	If goes into alarm the worker will take the necessary action as depicted in the local rules.	■ Surface contamination of workbench, equipment, floor and controlled areas ■ Monitoring of work clothing, body surface and small items of work when leaving a controlled area ■ Monitoring of large items and waste before being exported ■ Radioactive substances of vehicles and pedestrians when entering and leaving the site

340. It should be noted that solid waste monitoring is covered within PCER Chapter 5 (Ref. 121) whilst environmental monitoring is out of scope of Step 4 of GDA (Ref. 117).

341. From information within table 8, several areas required clarification or further information from the RP.

342. Clarification was required on what radionuclides are sampled regarding the gaseous waste system. The RP provided further information in the response to RQ-UKHPR1000-1660 (Ref. 20), whereby the RP stated to check radionuclide emissions from the reactor are compliant, the gas is sampled within the main stack via a common single point shrouded nozzle. Gas sampling is monitored for the following:

- Aerosols sampled via a filter paper to assess the radioactive concentration of alpha and beta emitted by the particulate matter.
- Iodine will be absorbed by the activated charcoal whereby the gamma emitted is detected.
- Noble gases is assessed for the concentration via alpha and beta emissions.
- ^{3}H, and Carbon-14 are periodically sampled via being sent to the laboratory for analysis
- In addition to the above, there is an independent sampling line which collects samples of the aerosol, iodine and noble gases which are sent to the laboratory for analysis.

343. As this covers an array of radionuclides which would be expected to be analysed as well as having an independent system to sample them, I judge this to be appropriate.

344. The control of ^{3}H is an important aspect and further information was required relating to how ^{3}H is sampled and monitored within the generic UK HPR1000 design. From the response to RQ-UKHPR1000-1660 (Ref. 20), the RP stated ^{3}H is monitored and sampled in various locations around the generic UK HPR1000 design such as ^{3}H in:

- Gaseous effluent
- Air of BRX
- Liquid Effluents
- Reactor coolant

345. These are all the possible escape routes for ^{3}H within the generic UK HPR1000 design under normal operations and samples are taken at various positions. I judge this to be appropriate.

346. There are neutron / criticality monitoring systems installed near the reactor core to monitor activity levels. However, there is no neutron / criticality system installed near the spent fuel pool and the fuel storage facility due to the following safety features (Ref. 118):

- Underwater fuel storage racks are designed to maintain spent fuel sub critical via the racks flooded with borated water, solid neutron absorbers in fuel racks and geometric arrangement.
- Fuel dry storage racks are designed to maintain the fuel in sub-critical state via geometrical structure. As well as no water pipes within the facility so as to stop the potential for water leakage.

347. In addition to this there is no neutron monitoring system within the BRX apart from near the reactor core. The RP provided further information within the response to RQ-UKHPR1000-0851 (Ref. 20) stating, during normal operation there is no access to the containment building, and so it instead relies on engineered and management controls to restrict access in these areas. If in the event workers have to access the containment structure, they will take portable neutron monitoring equipment. As the RP

has stated there are controls in place to restrict access during normal operation and there are arrangements in place for workers to access the area where they will have portable monitoring equipment available, I judge this to be acceptable and no further action needs to be taken.

348. Regarding the installed gamma monitoring within the generic UK HPR1000 design the RP stated these are installed if it meets one of the following criteria (Ref. 117):

- Where radiation dose rates may increase rapidly, and no other indication device is installed.
- Where radiation dose rates may rise rapidly and require evacuation.
- Occasional high radiation dose rates may occur and have an influence on workers.
- External control operation by other workers may cause rapid increase of dose rate.
- Where workers may access under accident conditions.

349. I have noted that this mainly covers green and yellow zoned areas and not orange or red zones. Under normal operations there is no access to red zones, though it is still feasible for workers to access orange zones. The RP has stated within response to RQ-UKHPR1000-1660 (Ref. 20) if workers have to access orange areas they will require a portable gamma monitor, so if this goes into alarm they take the necessary action.

350. Although there are appropriate management controls in place to restrict access to orange and red zones (i.e. interlock system, safe system of work), it is still feasible that workers will need to access to these areas for non-routine operations. If there was an installed monitoring system it would provide far better warning then a handheld monitor. As this is non-routine operation rather than a normal circumstance. I judge this to be a minor shortfall.

351. Regarding calculating doses to personnel, the passive dosimeter is used to measure the personal accumulated dose equivalent of external exposure for the worker during their time in a controlled area. From an RQ-UKHPR1000-1660 response (Ref. 20) the RP has stated there are a number of widely available passive dosimeters that can be used to measure external exposure. The EPD on the other hand provides a real time dose equivalent rate as well as the accumulated dose equivalent of a worker during their time in a controlled area.

352. In addition, the RP state whole body counters will be used to measure internal exposure. However, it is unclear from the generic UK HPR1000 safety case (Ref. 117) as well as RQ-UKHPR1000-1660 response (Ref. 20) if both or one of the above methods are to be used at the UK HPR1000. As this will be for the licensee to decide no further action needs to be undertaken during Step 4 of GDA.

353. Regarding dose control to personnel, as workers will be required to wear an EPD, this is read by the dosimeter readers which are situated at the entrance and exit of controlled areas. This information is sent to the electronic dose system.

354. When entering a controlled area this is achieved through the turnstiles whereby only authorised personnel may access. To achieve this, on the turnstile is an access card reader which transmits the information to the entrance dosimeter reader. Although this is not a requirement in the UK, it can be considered RGP by the RP.

355. The above illustrates a methodological approach to accessing a controlled area and an option for recording external exposure for workers, which also ensures that workers cannot access the controlled area without a dosimeter. This approach can also provide an additional control whereby an alarm threshold can be automatically programmed

into the EPD, which can be a useful tool to restrict exposure and ensure that personnel dose limits are not exceeded.

356. For surface contamination there are portable and installed contamination monitoring equipment. For the portable surface contamination equipment this is mainly used for all of the above aspects, whilst for installed monitoring equipment there is an array of equipment available (Ref. 117):

- Hand and foot monitors
- Vehicle gamma monitors
- Pedestrian gamma monitors
- Small item monitors
- Whole body surface contamination

357. Workers will be issued with portable gamma dose rate monitors (as well as neutron monitors where appropriate) to provide information for radiological protection purposes in addition to installed area dose rate monitoring equipment. If the alarm on these systems activate the workers will take the necessary action (Ref. 118). The approach undertaken by the RP regarding surface contamination monitoring for designated areas is in line with IRR17 regulation 19(10) (Ref. 10), hence I judge this to be acceptable.

358. Although not mentioned within the 'Radiation and Contamination Monitoring Topic Report' (Ref. 117), within the design manual (Ref. 120) it provides an overview of the EMIT approach to radiation and contamination monitoring equipment. It is noted that these aspects will be further developed during site-specific stage when definite monitoring equipment has been selected. The EMIT approach taken by the RP is in line with IRR17 regulation 20(3) (Ref. 10). I judge this to be appropriate.

359. From the information provided, the RP has adequately shown the generic UK HPR1000 design has an array of monitoring systems which provide protection to workers in designated areas that aligns with IRR17 regulation 20 requirements (Ref. 10). I judge the monitoring systems to be adequate.

4.7.1.2 Functional Design

360. As part of my Step 4 of GDA Radiological Protection assessment, I reviewed the functional design reports for the following monitoring systems:

- Area dose rate monitoring
- Area airborne activity monitoring
- Accident and post-accident monitoring.

361. For each of the functional design reports, the RP provided further information on the installed monitoring equipment as follows: (Ref. 118):

- Safety function categorisation
- Measuring task, object, method and range
- Operational conditions
- Alarm levels and alarm actions

362. The measuring method for area dose rate monitoring is gamma, whilst for area airborne activity as well as accident and post-accident is both beta and gamma. I raised a query within RQ-UKHPR1000-1610 (Ref. 20), to ask if monitoring is carried out for alpha. The RP responded stating that alpha in air levels are low and hence are not required to be continuously monitored by installed equipment. The RP went on to state alpha particles are monitored through portable monitoring equipment for surface contamination, as well as portable air monitors such as continuous airborne monitors.

As the RP has shown that alpha particles could be monitored by portable equipment which workers will carry when undertaking work in a designated area, I judge this to be adequate.

363. For the measuring range the RP has provided a number of principles (Ref. 118, Ref. 119) which the installed monitoring equipment must abide by including:

- Providing the minimum Gamma dose rate for when monitoring is required.
- The maximum is calculated on the design value source term or accident source term if the gamma radiation monitor is for post-accident monitoring.
- The maximum area airborne is calculated using the design value source term whilst the minimum is the realistic value source term.
- To account for possible measurement error of monitoring equipment the RP calculated a minimum and maximum measurement threshold.

364. I raised a query (RQ-UKHPR1000-1609) (Ref. 20) relating to the measurement error range stipulated within the above principles. In response to the RQ, the RP stated through their own assessment of guidance such as the 'Nuclear power plants – Instrumentation important to safety radiation monitoring for accident and post-accident condition's part 1: general requirements' (Ref. 122), there will be an intrinsic error of monitoring equipment (about 30%), hence the RP has been conservative with the measurement range. Although this could be seen as over conservative, one of the other principles is the current technical level of radiation monitoring equipment be considered which means monitoring equipment will be able to measure the ranges used. For this reason, I judge this to be acceptable.

365. The RP provided a worked example within the response to RQ-UKHPR1000-1609 (Ref. 20), of how the above principles are used to calculate the range. From reviewing the response, I judge the principles stipulated by the RP for calculating the measuring range to be appropriate.

366. Regarding design principles for alarms and actions to be undertaken, this varies dependent on the monitoring system, whereby:

- There are a number of alarms for installed gamma monitors, one is to inform workers the dose rate is abnormal whilst another is to state the dose rate is above the upper limit of the radiation zone. For gamma monitors which are in a green zone such as service corridors, an alarm is used to warn workers not to stay in the area. There is no automatic locking action for these types of monitors.
- The gamma dose rate monitoring channels above the SFP and reactor pool are set up with multiple alarm levels. As the alarm level increases this warns workers that working time is reduced to the point that they must evacuate the area.
- For area airborne radiation monitoring the alarm is activated when mitigating measures are required (for example an automated purification system). In some areas there are several alarms, whereby one is an early warning whilst another is when mitigating measures are required.
- For the MCR air contamination system, an alarm indicates the MCR air intake is polluted with radioactive particles and switches the air intake to iodine filtration automatically. Another alarm informs the operators within the MCR the air intake is heavily polluted with airborne radioactive contamination and reminds operators to manually switch air supply. I raised RQ-UKHPR1000-1609 (Ref. 20) relating to this claim. I judged the RP response to be adequate.

367. For all installed monitoring equipment when the dose in an area exceeds the limits specified, there are audible and visual alarms which will activate locally to warn

workers in the region to take the necessary steps. The alarms are also displayed within the MCR simultaneously.

368. As stated above for accident and post-accident monitoring there are no local alarms which activate. I queried this within RQ-UKHPR1000-1609 (Ref. 20), whereby the RP stated that this monitoring equipment is used after an accident has occurred hence workers should not be in the area and alarms will not be required.

369. I also noted that some installed monitoring equipment has no automatic actions, instead it will be up to the worker to follow the alarm manual procedures. I raised a query for the RP to justify this approach within RQ-UKHPR1000-1609 (Ref. 20). The RP response stated when the alarm activates this provides information to the MCR, enabling the operators to take the necessary action as stipulated within the alarm manual. In addition, radiological protection staff take the necessary management action. The alarm manual and management arrangements will be decided at site licensing.

370. From the information sampled regarding the functionality of the monitoring equipment, the RP has provided adequate evidence whereby:

- The methodology for measuring method and range is adequate.
- The design principles for alarms are adequate.
- Installed monitoring equipment provides information to workers on site as well as the MCR.
- Appropriate alarm actions are undertaken whether remote or by the operators.

371. With the above in mind, I judge the generic UK HPR1000 design has adequate monitoring functions which are in compliance with the relevant requirements of IRR 17 Regulation 20 (Ref. 10).

4.7.1.3 ALARP Assessment of Radiation and Contamination Monitoring

372. The RP has followed a systematic approach for the ALARP assessment of radiation and contamination monitoring, as can be seen in figure 6.

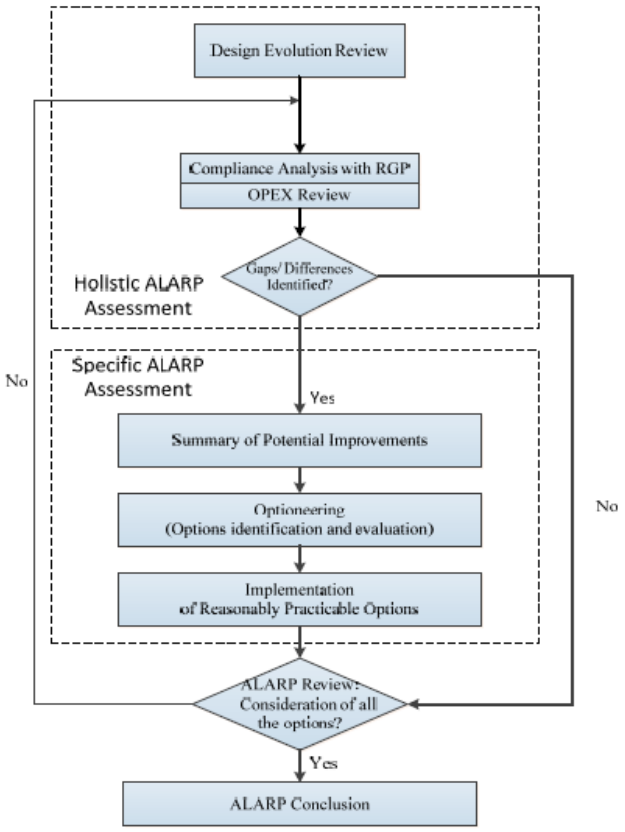

Figure 6: Provides the logic diagram to consider if radiation and contamination monitoring is ALARP (Ref. 117).

373. The first stage was the evolution review of the generic UK HPR1000 design. The RP provided information through OPEX on how the older generation of reactors (CPR1000, ACPR1000) influenced the generic UK HPR1000 design. Information has been used to improve the radiological and contamination monitoring to help reduce exposure to workers and members of the public. From this review, improvements for the generic UK HPR1000 design were identified in the following areas:

- Process activity monitoring
- Area dose rate monitoring
- Area Airborne monitoring
- Accident and post-accident monitoring

374. The RP also reviewed the radiological and contamination monitoring design to current guidance to identify any gaps in compliance. As the Chinese regulations are based on the IAEA standards and ICRP publication 103 (Ref. 123) the generic UK HPR1000 design is consistent with these standards. The RP has also undertaken an additional review against IRR17 (Ref. 10). Information relating to these reviews is detailed within the 'Radiation and Contamination Monitoring Topic Report' (Ref. 117). From the RPs assessment they concluded the generic UK HPR1000 design is compliant with current standards and guidance. From my assessment of the radiation and contamination monitoring, I judge this to be adequate.

375. The next stage undertaken by the RP was to review relevant OPEX from the following sources of information:

- Lessons learnt in previous GDAs
- Advance PWR design features
- CGN OPEX feedback

376. The RP provided evidence to corroborate that an OPEX review has been undertaken using this information where appropriate for the generic UK HPR1000 design. The RP state that no gaps were identified from the review (Ref. 117).

377. The next step would be using the ERICPPE methodology, however as monitoring is one of the tools for control, a different approach needs to be undertaken. Hence the RP has taken the following approach:

- The applicable codes and standards are reviewed for any omissions.
- Review of OPEX feedback to ascertain if this has been implemented in the design.
- Check if there are better options currently available which have not been implemented within the design.

378. As stated above, the first two aspects were reviewed in the holistic ALARP assessment. Regarding the third aspect, from reviewing potential options to improve the generic UK HPR1000 design, one improvement was identified regarding the use of a fuel cladding failure monitoring system. This will be explored further during site-specific stages.

379. From the review of the ALARP assessment of radiation and contamination monitoring, the RP undertook an appropriate methodological approach. The RP also provided appropriate evidence to corroborate the claim and arguments, that RGP and OPEX from previous reactor designs, as well as current standards have been considered within the radiological and contamination monitoring design, for the generic UK HPR1000 design.

380. In summary, I judge the RP has provided an appropriate ALARP argument that the radiation and contamination monitoring for the generic UK HPR1000 design is adequate.

4.7.2 Strengths

381. From the above assessment on radiation and contamination monitoring as outlined in Section 4.7 of my assessment, the following strengths from the RP generic UK HPR1000 safety case are:

- The RP has provided appropriate information to meet the expectations of current standards and guidance in relation to radiation and contamination monitoring.
- The radiation and contamination monitoring system for the generic UK HPR1000 design cover an array of different processes.
- An appropriate ALARP methodology has been undertaken.
- Functional reports sampled are in line with expectations for step 4 of GDA.

4.7.3 Outcomes

382. I have identified one minor shortfall within this section.

383. The radiation and contamination monitoring arrangements for the generic UK HPR1000 design are adequate.

4.7.4 Conclusion

384. Based on the outcome of my Step 4 of GDA Radiological protection assessment of radiation and contamination monitoring, I have concluded:

- The generic UK HPR1000 design has an array of monitoring systems which provide protection to workers in designated areas that aligns with IRR17 Regulation 20 requirements (Ref. 10).
- The generic UK HPR1000 design has adequate monitoring functions which are in alignment with IRR 17 Regulation 20 (Ref. 10).
- An appropriate ALARP argument has been provided that the radiation and contamination monitoring for the generic UK HPR1000 safety case is adequate.

4.8 Post-Accident Accessibility

4.8.1 Assessment

385. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I assessed the doses to workers employed in post-accident response and the demonstration that these doses are ALARP and compliant with legislative requirements.

386. Notwithstanding the overriding principle of restriction of radiation exposure to ALARP, in the event of a radiation emergency, Regulation 19 of REPPIR19 (Ref. 11) stipulates that the dose limits in IRR17 can be disapplied for authorised emergency workers, and reference levels in Regulation 20 can be applied in accordance with the emergency plan. Within REPPIR (Ref. 11) it specifies that the operator must ensure that the emergency plan prioritises keeping effective dose below a 100mSv reference level (this equates to DBA and Design Extension Condition A (DEC-A)), whilst in exceptional circumstances in order to save life the reference level may be raised up to but not exceeding 500 mSv (Severe Accidents (SA)).

387. Within the PCSR Chapter 22 (Ref. 3), the RP has the following sub-claim:

*"*The risk to workers mitigating fault/accident conditions complies with UK legal requirements and is ALARP*"*

388. This sub-claim is part of the overarching claim which encompasses workers and members of the public relating to ionising radiation from fault and accident conditions. The RP provide an argument and evidence to corroborate this claim through PCSR Chapter 22 Section 22.11 (Ref. 3), whereby the RP provides an overview of the following:

- Methodology for PAA and PAA requirements
- Dose assessment results for intervention workers
- ALARP demonstration

389. Further information is provided in the following documents which were assessed:

- 'Post-Accident Accessibility Topic Report' (Ref. 124)
- 'Worker Dose Assessment Methodology for DBA' (Ref. 125)
- 'Worker Dose Assessment for Representative DBA' (Ref. 126)

390. From the initial assessment of the above documents further information was required, hence RQs were raised (Ref. 20):

- RQ-UKHPR1000-1444
- RQ-UKHPR1000-1702
- RQ-UKHPR1000-1713
- RQ-UKHPR1000-1744

391. In addition, within this section I will discuss the emergency arrangements for the generic UK HPR1000 design in Section 4.8.1.6.

4.8.1.1 Methodology and Assumptions of PAA Analysis

392. The methodology and assumptions used by the RP, to determine what aspects are classed as DBA, and how doses are calculated are provided within:

- 'Worker Dose Assessment Methodology for DBA' (Ref. 125)
- 'Worker Dose Assessment for Representative DBA' (Ref. 126)

393. In total 13 DBA representative accidents have been considered by the RP. Further information on these events are provided in Section 4.8.1.2.

394. Regarding calculating the dose to workers during an incident, the RP has categorised workers into the following groups to help with worker dose assessment (Ref. 125):

- Workers in the MCR
- Local accident intervention workers
- Accidentally involved workers
- Other workers

395. Within this report I will assess the dose to workers for local accident intervention workers as well as workers within the MCR. For workers affected by an accident, and others, this is assessed within GDA Step 4 Fault Studies Assessment report (ONR-NR-AR-21-014) (Ref. 127), whilst for severe accidents this is GDA Step 4 Severe Accident Analysis Assessment Report (ONR-NR-AR-21-008) (Ref. 128).

396. For the above type of workers, the following exposure routes have been determined. Dependent on the worker category the exposure routes will vary due to (Ref. 125):

- Direct radiation from the source
- External exposure by submersion of airborne radioactivity
- Internal exposure by inhalation of airborne radioactivity

397. The source term used for a radiological emergency is the accident source term, whereby two types of sources are considered (Ref. 125):

- Contained source in critical location
- Unsealed sources released to the environment

398. Using the accident source term the RP undertook computer modelling codes to calculate the direct exposure. For airborne contamination this was calculated using dose conversion factors and empirical formula. It should be noted the references used for conversion factors are not the latest versions available. However through work undertaken reviewing HSG253 (see Section 4.4.2) it was noted the differences between the latest guidance and ones used by the RP are marginal, and I judged to be acceptable, though it is preferred to use the latest codes. I judged this to be a minor shortfall.

399. When calculating the potential dose to a worker several variables need to be considered (Ref. 126):

- Location of the worker relative to the radiological emergency
- Exposure time of the worker
- Effect of any PPE

400. The local accident intervention workers will have the largest potential exposure due to the role of resolving the incident. The RP has made the following assumptions when calculating their dose (Ref. 124):

- Operation time to resolve an accident is conservatively assumed based on OPEX information from CGN.
- A protection factor of 1000 for aerosols from respiratory devices is taken into account regarding inhalation. This is based on the 'UK Nuclear industry good practice guide to Respiratory Protective Equipment' (Ref. 129).
- The speed of the person is considered 3.5 km/h (lower than normal walking speed) due to potential environmental issues.
- The access route is the shortest distance between operations and destination, and the time there and back is the same.

401. For the following scenarios, the RP assumed it will take 30 minutes to resolve the issue if this involved closing valves. This was raised within RQ-UKHPR1000-1444 (Ref. 20), whereby the RP provided further evidence to clarify their claim. The RP stated that from OPEX and discussion with commissioning and operational departments, there were a number of variables to be considered (e.g. valve type, valve stroke, environmental factors) that would have to be accounted for. Hence the RP deemed a response of 30 minutes to be conservative. I judge this to be acceptable.

402. From information provided, it is unclear if the respiratory protection factor used by the RP is suitable for all potential radionuclides following an airborne release radioactivity. From my interpretation of the 'UK Nuclear industry good practice guide to Respiratory Protective Equipment' (Ref. 129), the protection factor stated is suitable for some radionuclides. This does not apply to all radionuclides (such as ^{3}H and noble gases), where the assumption of a respiratory protection factor of 1000 is not appropriate. Calculations undertaken by the RP using the protection factor of 1000 may underestimate potential internal radiation exposure to local intervention worker for certain scenarios. Though this is likely to be a relatively small component of the total radiation exposure to workers during an incident, there are aspects which require further work and I judge this to be a finding rather than a shortfall. Consequently, I have raised the following AF:

AF-UKHPR1000-0105 – The licensee shall during detailed design, demonstrate local intervention worker internal radiation exposures are reduced so far as is reasonably practicable. Justification should be provided that the respiratory protection factor of 1000, used to calculate internal radiation exposure, is appropriate. This is required where the safety case identifies the necessity for respiratory protective equipment to be worn during/following accidents, which includes: • design basis accidents; • design extension condition A: and • severe accidents.

403. Notwithstanding the above AF, from assessing the RP methodology and other assumptions made, I judge those to be adequate from my experience of emergency preparedness and response.

4.8.1.2 DBA Assessment

404. As stated in above sub-section, the RP has identified 13 representative DBAs, these are (Ref. 124):

- Turbine Trip
- Steam Generator Tube Rupture (one tube)
- Small break (Loss Of Cooling Accident (LOCA)
- Rupture of a line carrying primary coolant outside containment
- RCV[CVCS], volume control tank rupture

- Spectrum of Rod Cluster Control Assembly (RCCA) ejection accident
- Steam system piping large break
- Large Break LOCA
- Steam Generator Tube Rupture (two tubes in one steam generator)
- Residual Heat Removal System piping break inside containment
- Reactor coolant pump seizure or reactor coolant pump shaft break
- Dropping of fuel assembly
- Dropping of spent fuel cask

405. The above DBAs can be resolved via MCR operations or local intervention as a worst case scenario (local intervention only occurs if the MCR system has failed).

406. The calculation and results for the local intervention workers are provided within the PAA topic report (Ref. 124) and in my judgement are reasonable.

407. From the results for local intervention worker 8 out of the above 13 DBAs require local intervention action with the highest exposure calculated as 36 mSv, which is for spectrum of RCCA ejection accident (Ref. 124). This is below the 100 mSv reference level stipulated within REPPIR19 (Ref. 11). I judge this to be reasonable and from what I have seen, doses appear to have been reduced SFAIRP from a GDA perspective.

408. Regarding dose to workers within the MCR from the above DBA, the RP provided information within RQ-UKHPR1000-1444 (Ref. 20). The highest calculated dose is 4.65 mSv which is from spectrum of RCCA ejection accident which is below the 100 mSv reference level stipulated within REPPIR19 (Ref. 11). I judge this to be reasonable and from what I have seen, doses appear to have been reduced SFAIRP from a GDA perspective.

4.8.1.3 DEC-A Assessment

409. DEC-A are faults considered to be beyond DBA. The RP has reviewed the potential DEC-A and identified eight sequences which require local intervention where an operator will receive an exposure. These sequences are (Ref. 124):

- Loss of three fuel pool cooling and treatment systems
- Loss of ultimate heat sink for 100 hours
- Small Break LOCA with total loss of low head safety injection
- Station black out in shutdown condition
- Station black out
- Station black out for spent fuel pool
- Total loss of colling chain in shutdown state
- Total loss of colling chain with reactor coolant pump sealing leakage

410. The assumptions as stated in Section 4.8.1.1 have been used to calculate the dose to workers for DEC-A.

411. From reviewing the results, the dose to local intervention workers for completing the necessary actions for the above sequences range from 1.84 mSv to a 26.5 mSv (Ref. 124). This is below the 100 mSv reference level stipulated within REPPIR19 (Ref. 11). I judge this to be reasonable and from what I have seen, doses appear to have been reduced SFAIRP from a GDA perspective.

4.8.1.4 SA Assessment

412. For severe accident assessment, the RP identified two key systems where local intervention is required. These systems are (Ref. 124):

- Containment Filtration and exhaust system

- In-Vessel Retention

413. The assumptions as stated in Section 4.8.1.1 have been used to calculate the dose to workers for severe accidents.

414. From reviewing the results, the dose to local intervention workers for completing the necessary actions for the above sequences range from 26 mSv to 229 mSv (Ref. 124). This is below the reference level for severe accidents which is up to but not exceeding 500 mSv as stipulated within REPPIR19 (Ref. 11). I judge this to be reasonable and from what I have seen, doses appear to have been reduced SFAIRP from a GDA perspective.

415. Regarding the dose to workers in the MCR following a severe accident, the RP provided information in response to RQ-UKHPR1000-1713 (Ref. 20), whereby they stated the methodology and calculations for the above severe accidents. The maximum effective dose over a 30-day period is 37 mSv. This is below the reference level for severe accidents which is up to but not exceeding 500 mSv as stipulated within REPPIR19 (Ref. 11). I judge this to be reasonable and from what I have seen, doses appear to have been reduced SFAIRP from a GDA perspective.

4.8.1.5 ALARP Assessment of PAA

416. The RP followed the ALARP methodology, that has been implemented for the generic UK HPR1000 design aspects to demonstrate PAA worker dose has been reduced SFAIRP.

417. For the holistic ALARP assessment the RP has reviewed OPEX from previous designs of Chinese reactors as well as lessons learnt from previous GDA assessment, to identify any shortfalls in relation to standards and guidance. From the holistic assessment the RP stated no shortfalls were identified (Ref. 124).

418. For the specific ALARP assessment the ERICPPE approach has been undertaken. I noted that the RP had eliminated where possible the need for manual tasks and replaced these by remote operations via the MCR for DBA accidents. Manual operations are only required if unable to be completed from the MCR.

419. The RP stated for DEC-A and SA events, local operation is required and cannot be completed remotely from the MCR. The RP has justified this action due to the low frequency of these events occurring. Although these cannot be remotely operated from the MCR, the RP has installed remote operation such as the open and close of containment filter and exhaust system where possible. This is to replace near-distance operation (Ref. 124) to minimise dose to workers. I judge this to be acceptable.

420. Regarding reduction of potential dose, the RP has considered human factors and ergonomics in the event of an emergency, as well as worker training being undertaken for practice and familiarisation. In addition, the shortest route to an incident has been chosen to reduce the time workers are exposed (Ref. 124).

421. For isolation aspects the RP adopted shielding (such as bulk shielding design) or containment to isolate radioactive sources. For control and PPE aspects this is reliant on administrative controls (access control, risk assessments) and the appropriate PPE as well as RPE during PAA (Ref. 124). These aspects will be determined at site-specific stages, I judge this to be reasonable.

422. The above measures were considered by the RP when calculating the dose to local intervention workers in the event of an accident.

423. Regarding dose to workers within the MCR, as stated in DBA assessment the doses are reasonable as they are significantly below the reference level of 100 mSv as stipulated within REPPIR19 (Ref. 11). From what I have seen these doses appear to have been reduced SFAIRP from a GDA perspective. The RP provided additional information of the ALARP measures in response to RQ-UKHPR1000-1444 (Ref. 20) in relation to mitigating dose to operators within the MCR, whereby the MCR has:

- A positive pressure to minimise inward leakage
- Aerosol and iodine filters are attached to the ventilation system
- Air intake measured by gamma monitors whereby if they go into alarm the iodine filtration unit is activated
- Radiation shielding due to the concrete walls around the MCR.

424. From the review of the ALARP assessment, the RP has undertaken a methodological approach to have appropriate mitigating measures to reduce exposure to local accident intervention workers. Where practicable the RP has introduced remote operations to eliminate workers needing to undertake manual operations. In addition, all doses to local intervention workers are below the reference levels as stipulated in REPPIR19 (Ref. 11). Hence in my judgement the RP has provided an adequate ALARP argument for local intervention workers.

425. Regarding the ALARP argument to operators in the MCR, the ALARP methodology is appropriate with suitable measures in place to reduce exposure in the case of an accident. The highest calculated dose was below 5 mSv for a DBA.

426. Hence in my judgement the RP has provided appropriate argument and evidence to corroborate the claim that risk to workers mitigating fault/accident conditions complies with UK legal requirements and is ALARP.

4.8.1.6 Emergency Arrangements

427. The RP has PCSR Chapter 32 on emergency arrangements (Ref. 18). The overarching sub-claim made by the RP is the following:

"Emergency arrangements will be in place, prior to commissioning, that will be in accordance with up-to-date standards in the event of a release of radioactive substances".

428. It should be noted that evidence to corroborate the claim and arguments provided within the chapter are not available during Step 4 of GDA and will only be available during site-specific stages. Instead, the RP provided information on what emergency arrangements will be in place for the generic UK HPR1000 design, and assessed these aspects against the current standards. If the RP identified a shortfall, further information will be provided.

429. There are a range of applicable codes and standards that relate to emergency arrangements. Within the UK, REPPIR19 (Ref. 11) provides the legal framework for emergency arrangements for the protection of workers and the public from a radiation emergency, and ensures that relevant members of the public are supplied with prior information specified in REPPIR19. Within REPPIR 19 (Ref. 11) the licensee is responsible for on-site emergency arrangements, though off-site emergency arrangements are the responsibility of the local authorities. Some responsibilities of the licensee are as follows:

- Evaluation of the hazards associated with the site to determine if it has the potential to cause a radiological emergency;
- Preparation of an emergency plan;
- Review and testing of on-site emergency planning; and

- Provision of suitable training to operators so they are SQEP regarding emergency arrangements.

430. The RP has undertaken a review of the applicable codes and standards to ascertain if the design of the generic UK HPR1000 design is compliant, most notably the RP has based the arguments on IAEA SSG-25 specifically safety factor 13 (Ref. 130) which relates to emergency arrangements for NPP.

431. The RP has provided information on aspects which are likely to be included in the on-site emergency plan for the generic UK HPR1000 safety case. I judge this to be in line with current standards and guidance.

432. The RP provides a list of on-site emergency response facilities which will form part of the generic UK HPR1000 safety case emergency plan. These facilities are (Ref. 18):

- MCR – This is the workplace for operational staff and fulfils two roles (provide control measures to maintain or restore the safety status of the generic UK HPR1000 design, as well as serve as major facility for emergency response command).
- Remote shutdown station – This is to ensure the reactor is placed and maintained in a shutdown state and monitored for any issues. If the MCR becomes defunct the remote shutdown station can take over.
- On-site emergency control centre – This is the strategic centre to control and co-ordinate the emergency response for both on and off site.
- Technical support centre – This is operational during an emergency where the technical support group initiate and implement emergency response actions. This is situated above the MCR.
- Operation support centre – Serves as the workplace for the staff of the maintenance and administration/logistics during an emergency response.
- Public information centre – This is to communicate with the public and news media to inform them of any changes regarding the emergency incident. In addition the centre can hold press conferences.

433. The RP provided additional information on the habitability of the MCR (which includes the technical support centre within this envelope) through the response to RQ-UKHPR1000-1744 (Ref. 20). The MCR has the following radiological protection measures:

- Filtering the gaseous radioactive materials.
- Reduction of the radioactive leakage in the MCR envelope.
- Radiation shielding of the building walls.
- Provision of individual personal protection equipment.

434. It should be noted that the habitability of other buildings during an emergency other than the MCR and technical support room, are out of scope of Step 4 of GDA and will be reviewed during site-specific stages. I judge this to be appropriate.

435. The communication system during an emergency are provided for reliable and efficient communication, with an effective and diverse means of communication during normal operation and emergency (Ref. 18). The RP provided further information through the response to RQ-UKHPR1000-1744 (Ref. 20) on the type of communication systems which will be available:

- Primary and secondary telephone system
- Public address system
- Alarm system
- Wireless communication system

436. The RP has provided the strategy for on-site accident management through the use of the emergency operating procedures (EOP), and severe accident management guidelines (SAMG).

437. EOP are used to support and guide operators in preventing and managing incidents. Within the above section (4.8.1.2) information is provided on the possible accidents which have a radiological protection aspect. EOP for the site will be developed during site-specific stages (Ref. 18). I judge this to be acceptable.

438. In response to RQ-UKHPR1000-1744 (Ref. 20), the RP provided further information on SAMG. This was guidance designed to mitigate severe accidents, using not only dedicated severe accident mitigation systems, but also other available equipment, which may not be credited under severe accident (Ref. 20). This will be completed during site-specific stages, I judge this to be acceptable.

439. A high-level ALARP assessment was undertaken by the RP to review specific RGP from different standards and guidance. Within the PCSR Chapter 32 (Ref. 18) the RP identified no gaps.

440. In summary the RP has provided high-level information of the emergency arrangements that will be in place for the generic UK HPR1000 design and have shown there are no gaps with current standards and guidance. As stated earlier, only claims and arguments have been presented with evidence to be provided at site-specific stages.

441. As stated within this section of the report, significant information relating to emergency arrangements will be completed as normal business during site-specific stages in accordance with License Condition (LC) 11 – Emergency Arrangements, under LC11(1) (Ref. 131):

"Without prejudice to any other requirements of the conditions attached to the licence the licensee shall make and implement adequate arrangements for dealing with any accident or emergency arising on site and their effects"

442. In my judgement, the information provided by the RP regarding emergency arrangement is adequate for Step 4 of GDA.

4.8.2 Strengths

443. From the above assessment on PAA as outlined in Section 4.8 of my assessment, the following strengths from the RP safety case for the generic UK HPR1000 safety case are:

- The RP has provided an appropriate methodology in determining the dose to local intervention worker as well as workers within the MCR for DBA, DEC-A and SA scenarios.
- The doses calculated to workers for the DBA, DEC-A and SA are all below the respective reference levels as stipulated within REPPIR19 (Ref. 11) and from what I have seen appear to have been reduced SFAIRP from a GDA perspective.
- The RP has followed an appropriate ALARP methodology.
- The RP where practicable have eliminated the need for a worker to manually activate contingency arrangements, instead utilising a remote approach from the MCR.
- The RP has a good understanding of what is required for emergency arrangements for the generic UK HPR1000 design if built within the UK.

4.8.3 Outcomes

444. The doses calculated for the representative DBA, DEC-A and SA are below the respective reference levels under REPPIR19, for local intervention workers and MCR workers. From what I have seen I am also satisfied that doses have been reduced SFAIRP from a GDA perspective.

445. I have identified the following AF:

- AF-UKHPR1000-0105 – The licensee shall during detailed design, demonstrate local intervention worker internal radiation exposures are reduced so far as is reasonably practicable. Justification should be provided that the respiratory protection factor of 1000, used to calculate internal radiation exposure, is appropriate. This is required where the safety case identifies the necessity for respiratory protective equipment to be worn during/following accidents, which includes:
  - design basis accidents;
  - design extension condition A; and
  - severe accidents.

446. I have identified no minor shortfalls in relation to PAA and emergency arrangements for the generic UK HPR1000 design.

4.8.4 Conclusion

447. Based on the outcome of my assessment of PAA, I have concluded that overall, the approach undertaken by the RP is reasonable and fulfils the sub-claim stipulated within the PCSR Chapter 22 (Ref. 3).

448. Whilst I have identified an AF in my assessment, it is my opinion that this can be resolved by the licensee during site-specific stages.

449. Based on the outcome of my assessment of emergency arrangements, I have concluded the information provided by the RP is acceptable for Step 4 of GDA and fulfils the sub-claim stipulated within the PCSR Chapter 32 (Ref. 18).

4.9 Criticality Safety Assessment of the New Fuel Store and Spent Fuel Pool

4.9.1 Assessment

450. Within the PCSR Chapter 22 (Ref. 3), the RP has made the following claim:

"The risk to workers and members of the public from the potential harmful effects of ionising radiation resulting from fault and accident conditions complies with UK legal requirements and is ALARP".

451. This claim encompasses workers and members of the public whereby during a fault / accident condition the risk from ionising radiation is restricted in line with current guidance / UK legal requirements. My assessment reviews the arguments and evidence to corroborate this claim with regard to potential criticality accidents.

452. Essential to the long-term operation of the reactor are two fuel stores – the NFA Store and the SFA Pool. I shall provide only a superficial description of these stores; interested parties requiring greater detail can consult the relevant sources (Ref. 132).

453. The NFA Store – Up to twelve NFAs (i.e. unirradiated FAs), in two rows of six, will be held in stainless steel racks until required in the reactor. The store will be air-filled. A description in terms of dimensions and materials, suitable for criticality-safety modelling

is available in the relevant sources (Ref. 132). Schematics showing the layout used for the criticality calculations and modelling are shown below in Figure 7 and Figure 8:

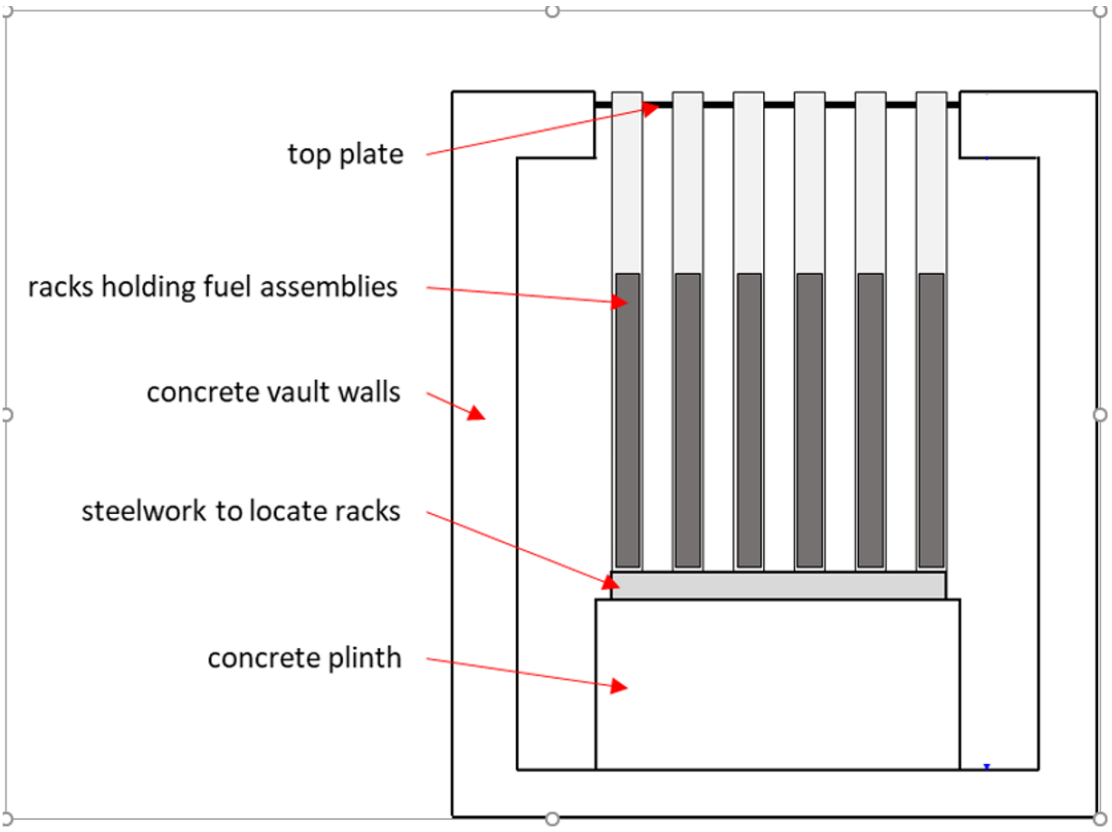

Figure 7: General Arrangement of NFA Store (Vertical Slice)

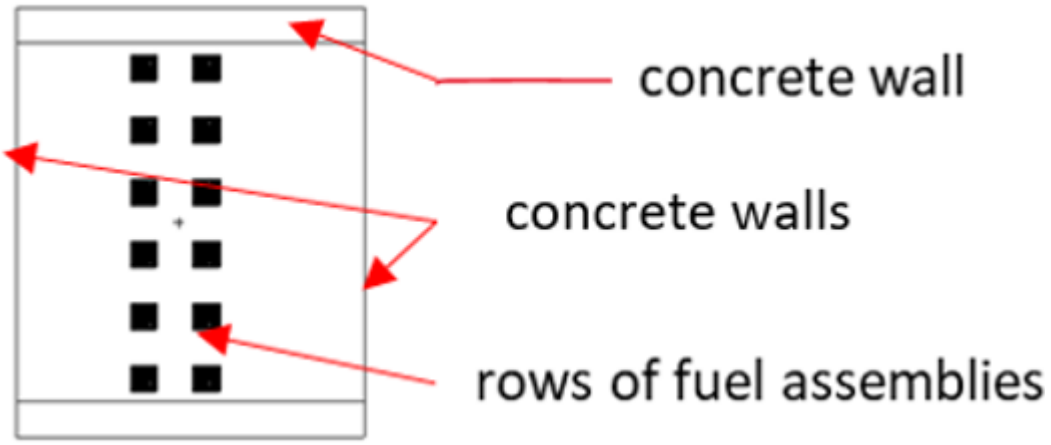

Figure 8: General Arrangement of NFA Store (Horizontal Slice)

454. The SFA Pool – Up to 49 SFAs (i.e. irradiated FAs), will be held in 7x7 stainless-steel racks. The SFA Pool will be filled with boronated water. A rack may contain:

- New (i.e. unirradiated) FAs, awaiting transfer into the reactor core; and/or
- spent (i.e. irradiated) FAs, awaiting removal to a long-term disposal facility.

455. A schematic showing the layout used for the criticality calculations and modelling is shown below in Figure 9:

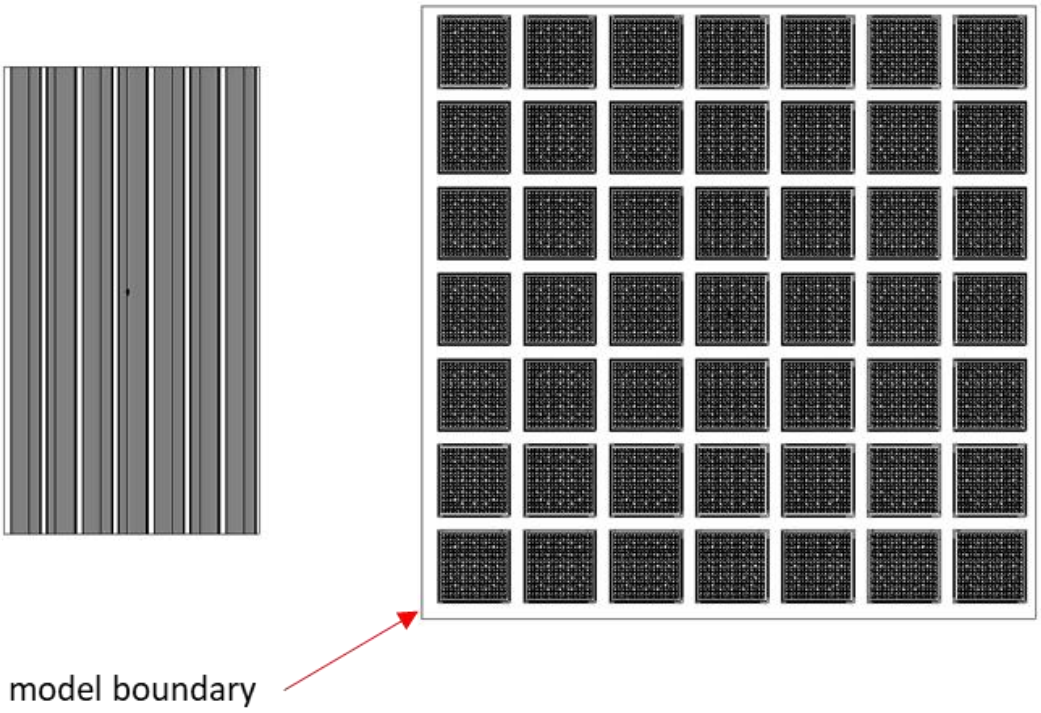

Figure 9: Vertical and horizontal sections through the SFA Pool

456. Neutron absorbing panels and soluble neutron poison will be used to help maintain criticality safety in the SFA Pool.

457. A description in terms of dimensions and materials suitable for criticality-safety modelling is available in the relevant source (Ref. 132) and responses to RQ-UKHPR1000-1077 and RQ-UKHPR1000-1381 (Ref. 20). The purpose of my assessment was to obtain confidence that the NFA Store and SFA Pool generic designs, if constructed, would be safely subcritical during normal operations and in the event, and as a result, of credible fault conditions (i.e. accidents, maloperations and extreme natural events). The criticality safety of the NFA Store and SFA Pool, which are designed to hold considerable quantities of fuel, is an important consideration for ensuring the safety of operational personnel and the public.

458. In the UK there is no specific legislation directly governing the requirements for maintaining criticality safety and, for nuclear site licence holders, regulation is carried out mainly under the conditions attached to the nuclear site license, the Health and Safety at Work Act 1974 (Ref. 133) and the IRR17 (Ref. 10).

459. I confined the scope of my Step 4 of GDA Criticality assessment, to the criticality safety of the NFA Store and SFA Pool to the proposed generic UK HPR1000 design. Loading FAs to the NFA Store and unloading from the SFA Pool, and ancillary operations with these, are out-of-scope (see Section 2.3) and have not been assessed.

460. My assessment was informed mainly by the ONR SAPs (Ref. 2), and the ONR TAG on criticality safety (NS-TAST-GD-041) (Ref. 4). Other ONR TAGs and international guidance were used, and these will be referenced in the text.

461. The SAPs which specifically apply to my criticality safety assessment of the NFA Store and SFA Pool are ECR.1, ECR.2 and FA.2; these are explained in Annex 1.

462. In my judgement, and consistent with the advice provided inNS-TAST-GD-041 (Ref. 4) and guidance in IAEA SSG-27 (Ref. 134), in general, the aspects of a criticality safety justification for a nuclear design which are most important to its adequacy are:

- The Criticality Safety Criteria (CSC). Appropriate criteria are necessary to define what can be considered to be safe from a criticality safety perspective. CSC are defined with reference to several considerations, for example: allowances for potential errors and uncertainties, and with a margin to allow for unforeseen and unforeseeable events (Ref. 135).
- The computer code(s) and nuclear data used to carry out the criticality calculations, and an understanding of their features, limitations, and accuracy in predicting neutron multiplication factors. The latter two factors can be highly dependent on the materials and arrangements being assessed. Therefore, an important part of this aspect is “Validation & Verification”, essential to provide confidence that the calculations and methods are sufficiently accurate and to calculate the maximum error or “bias”, for inclusion in the CSC (Ref. 136) (Ref. 137).
- The technical competence of the assessors and checkers. Criticality physics displays many non-intuitive behaviours (Ref. 138) and modern criticality codes are complicated and require specialised knowledge to use properly (see for example the ‘Monte-Carlo Criticality Calculations with MCNP6-Whisper Tutorials’, (Ref. 139)). It is considered good practice in the UK for organisations to define the competences required of criticality safety specialists and ensure that their staff meet and maintain these (Ref. 140).
- Quality assurance arrangements, referring to information management and checking procedures to provide confidence that the calculations and assessments have been carried out correctly (Ref. 141).
- Normal and Fault Analysis: an understanding of the designs to be assessed and the range of normal, and credible abnormal and fault conditions that they may be subject to (see SSG-27 (Ref. 134) and ONR SAPs FA.2 (Ref. 2)).
- Sensitivity analyses to ensure that credible variations in parameters and assumptions cannot undermine the safety arguments.
- Results of calculations: these must satisfy the CSC.

463. These considerations informed my strategy for sampling the RP’s submissions and my assessment of the RP’s criticality safety justification focussed on these areas; a summary of my approach is presented below.

464. In assessing the RP’s criticality-safety justification for the NFA Store and SFA Pool, I initially had difficulties in forming an opinion about these areas because at the time of the RP’s submission (Ref. 142):

- ONR was unfamiliar with the JMCT criticality code – it is not used within the UK. (As explained in more detail in Section 4.2.1.2, JMCT is a neutronics code for carrying out nuclear transport calculations (including criticality) and is maintained by the Chinese Academy of Sciences).
- The RP’s arrangements for assuring the competence of its criticality assessors were not known by ONR.
- The RP’s arrangements for maintaining Quality Assurance (QA) in respect of criticality calculations and assessments were not understood by ONR.

465. To assist in ONR’s assessment a TSC (Ref. 143) – was engaged to carry out independent calculations of the criticality safety of the generic designs of the NFA and SFA stores and also to provide an independent opinion of the RP’s criticality safety documentation. The TSC used the MCNP criticality code (Ref. 144) for calculations and the work is reported (Ref. 145).

466. The outputs of the TSC's work were used to inform my assessment of the adequacy of the RP's criticality safety justification of the NFA and SFA Pool designs; however, the TSC was not directly involved in making the regulatory decision.

467. My assessment of the RP's criticality safety documents was supplemented by monthly technical meetings with the RP to obtain clarification about the various technical matters described below; technical meeting minutes are recorded (Ref. 146). During these technical meetings, formal presentations were made by ONR to explain the gap to RGP and ONR's expectations (Ref. 147), and also by the RP to clarify aspects of its submission and the improvements that had been made to the criticality justification (Ref. 148). During this period, the RP's criticality safety case evolved from Revision B (Ref. 142) to Revision D (Ref. 149). My overall assessment of the adequacy of the criticality safety of the NFA Store and SFA Pool is based on Revision D.

468. The following sub-sections summarise my Step 4 of GDA Criticality assessment.

4.9.1.1 The RP's Criticality Safety Criteria

469. When assessing criticality safety using calculations which produce estimates of the neutron multiplication factors (K), it is conventional to judge the acceptability of the value of K against numerical criteria. In doing so, it is RGP to make allowances for statistical uncertainty, observed calculation biases, credible uncertainties and variations in modelling parameters. Specific guidance (Ref. 135) on these matters for criticality work in the UK has been formulated by the UK Working Party on Criticality (UK WPC), a long-established UK industry working group (Ref. 150). The RP proposed CSC derived from US standards (Ref. 151), which are not entirely consistent with the current advice provided by the UK WPC. However, in my judgement the differences are not significant for this project. In any case, the differences in CSC are academic since all the RP's results (confirmed by TSC calculations – see below in sub-section 4.9.1.6) are below 0.95 and satisfy both approaches.

470. The RP's methodology which explicitly models errors/uncertainties in all factors using Monte Calo simulation is, in my judgement, an appropriate approach to understanding the effects of uncertainty. Overall, in my judgement the RP's approach to defining a CSC is adequate.

4.9.1.2 Computer Code and Nuclear Data

471. The JMCT neutronics code is maintained by the Chinese Academy of Sciences, and although information is available on the internet (see for example '3-D Monte-Carlo Neutron-Photon Transport Code JMCT and Its Algorithm' (Ref. 152) or 'JMCT Monte-Carlo Simulation Analysis of BEAVRS and SG-III Shielding' (Ref. 153)), this is not enough to provide ONR with sufficient confidence in the code's abilities. At the time of writing, and to the best of my knowledge, there is no experience of using this code in UK industry that could be drawn upon. The reasons for not simply accepting a criticality code's results at face value are explained below – these apply to all criticality codes, not just JMCT.

472. Criticality codes are extremely complicated items of software relying on advanced mathematical and statistical techniques to give accurate results. Furthermore, they depend on extensive quantities of nuclear data, the individual items of which are obtained from many sources and which are subject to the uncertainties inherent in measurement and processing. For use in a criticality code the nuclear data must be further processed (e.g. interpolation; approximations to deal with thermalisation; adjustments for temperature).

473. Both codes and data are potential sources of error; for example, errors and issues have recently been identified in both MCNP and SCALE criticality codes – currently

two of the world's standard and most well-supported criticality codes (Ref. 154) (Ref. 155).

474. It follows that currently no criticality neutronics codes are 100% accurate – imprecision in nuclear data, approximations in representing physical processes and uncertainties in geometrical and material data can combine to produce errors and uncertainties in the computed values of the neutron multiplication factor. The errors and uncertainties can be highly system dependent. Also, Monte-Carlo codes, by their nature, always have statistical uncertainties associated with the results.

475. Because of these factors, the most practical way to demonstrate the accuracy and reliability of a criticality code/data combination is to compare the results of the predictions of neutron multiplication factors against accurate measurements taken from carefully executed experiments, commonly referred to as "benchmarks" – there is an industry standard database of these (Ref. 156). The idea is to select benchmarks which are neutronically similar to the application (i.e. NFA Store and SFA Pool) and model them with the selected code & nuclear data library. The accuracy of the code/data can be inferred from the results of the calculations.

476. In the field of criticality safety, this exercise is often referred to as V&V, and RGP is to carry this out for every application of the code because even small differences in moderation or dimensions can significantly affect the code's accuracy. Extensive advice on this subject is available (see for example 'Review of studies on criticality safety evaluation and criticality experiment methods' (Ref. 157) and 'Overview of Approaches Used to Determine Calculational Bias in Criticality Safety Assessment' (Ref. 158)), and an American National Standard (Ref. 136) on the subject is available. Simply put, the V&V process consists of the following steps:

- Select an adequate number of appropriate benchmarks similar to the application to be modelled (e.g. a fuel store).
- Model the benchmarks with a particular code/nuclear data/computer combination and record the values of the neutron multiplication factors (K_B).
- Compare the K_B with the experimentally measured values of the neutron multiplication factors (= K_{exp}) of the benchmarks.
- Record the differences between the two (= $K_{exp} – K_B$) and the statistical uncertainties associated with these.
- Employ appropriate statistical techniques to derive an allowance for the overall error & uncertainty due to nuclear data bias to feed into the CSC.

477. In my experience, weaknesses in V&V often centre around two areas:

- In selecting/obtaining appropriate benchmarks which are sufficiently similar (neutronically) to the applications. (Indeed, for many years, "neutronic similarity" was not formally defined and even today across the nuclear industry there is no uniform approach to benchmark selection.).
- Treatments of the differences $K_{exp} – K_B$ to derive a nuclear data bias.

478. In the V&V (Ref. 153) (Ref. 142) (Ref. 159) (Ref. 149), the RP has largely followed the advice in American Nuclear Standard Institute (ANSI) Standard 8.24 (Ref. 136).

479. To select appropriate benchmarks, the RP has used "professional judgement" based on matching observable parameters of the models (e.g. enrichment, moderation, pin diameter) to those of the benchmarks. Historically, this approach to benchmark selection has been widely used through the nuclear industry, but in my judgement, it has shortcomings:

- Firstly, it cannot provide a numerical measure of "nuclear similarity", i.e. just how well the benchmarks match the models; and

- secondly, it cannot identify if the models have specific sensitivities to particular energy regions of nuclear data.

480. For these reasons, internationally V&Vs in many modern criticality assessments supplement professional judgement with a quantified approach based on sensitivity/uncertainty (S/U) Analysis. S/U Analysis tools are available in modern codes such as MCNP and SCALE – for example see 'Monte-Carlo Criticality Calculations with MCNP6-Whisper Tutorials' (Ref. 139), 'Scale/TSUNAMI Sensitivity Data for ICSBEP Evaluations' (Ref. 160), 'Validation of criticality calculation for systems with damp MOX powders' (Ref. 161) (the theory is explained in 'Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation' (Ref. 162)).

481. In my judgment, for the specific task of benchmark selection, the RP's approach would have benefited from S/U Analysis. On the other hand, there are mitigating factors:

- In the designs of current UK fuel storage facilities S/U Analyses have not been used – indeed in the UK there are only a few examples of this approach being employed and then only for atypical systems.
- In criticality safety terms, the NFA Store and SFA Pool are standard systems. They contain well-moderated and unmoderated low-enrichment uranium respectively, and no novel features or unusual materials. PWR fuel stores and pools have been extensively modelled over the past decades and are well understood, with plenty of experimental evidence to support analyses. Therefore, I expect that the JMCT code and data would perform with a similar degree of accuracy to the other major criticality codes.
- Generally, modern criticality codes and nuclear data sets can predict K in moderated and unmoderated low-enrichment uranium systems well, typically with best-estimate bias values of about 0.005-0.01 in K (e.g. see Table 4 within 'Overview of Approaches Used to Determine Calculational Bias in Criticality Safety Assessment' (Ref. 158)). Simply put, the major modern criticality codes are accurate to about 1% in K; the claim on code/data accuracy is usually limited by the accuracy of the underlying benchmarks – typically 0.3% in K for these systems. The RP's reported value of 0.00012 (~ 0.012%) for the nuclear bias (see paragraph 34 of 'Criticality Analysis of Fuel Storage' (Ref. 149)) is small in comparison.
- Also, the RP makes generous allowances for the other potential sources of error and uncertainty, arriving at a value of 2500 pcm[§] (~ 0.025 in K or 2.5%) for the overall error and uncertainty in K. Additional considerations are that the calculations have been conservatively carried out (e.g. in enrichment) and that the safety margins for normal operations and all but the most extreme fault conditions are very large.
- As far as I am aware, the use of S/U analysis is not mandated in any criticality standards or advocated in any criticality guidance documents.

482. Overall, after considering all these factors, I consider that the RP's approach to V&V is adequate for the criticality justification of the NFA Store and SFA Pool.

4.9.1.3 Technical Competence of the Assessors and Checkers and Quality assurance Arrangements

483. The following factors have aspects which are in common and were assessed in parallel:

- Technical competence of the assessors and checkers

[§] *PCM – A "per cent mille" or pcm is one one-thousandth of a percent. It can be thought of as a "milli-percent". It is commonly used in …. nuclear reactor engineering (Ref. 176).*

- QA arrangements

484. As described above, the physical processes governing criticality safety are complicated and sometimes counter intuitive. Modelling fissile systems with computer codes requires highly specialised knowledge. For these reasons in the UK, it is considered RGP to define and monitor competencies for criticality assessors and QA checkers (Ref. 140) and, under ISO 9000 (Ref. 141), to have formal, robust QA procedures in place to ensure that calculations have been correctly carried out.

485. Early in the Step 4 of GDA Criticality assessment (Ref. 146), ONR asked the RP to explain its arrangements for training, checking and QA, with respect to criticality safety. The RP responded quickly and openly and described their systems for maintaining the competence of their “Reactor Design” engineers, which include:

- Mentoring new engineers
- Monthly testing
- An annual exam over-seen by the lead Criticality Safety Engineer

486. I inspected RP training records that had been “signed-off” to show that levels of competence had been achieved.

487. Also, the RP explained that most of the engineers involved in the reactor design had been involved in the project for at least three years. Most of the reviewers/checkers had at least eight years of experience. The RP applied formal procedures for QA checking calculations and reports. The RP’s responses are recorded in ‘The Supplement Analysis on Criticality Safety’ (Ref. 148).

488. I also formed the view, from technical meetings with the RP and from reading reports and discussing technical items, that the RP engineers were competent in criticality physics and modelling. In summary, I judge that the RP’s arrangements for ensuring competence and QA, for criticality safety purposes, are adequate.

4.9.1.4 Credible Abnormal and Fault Conditions

489. The RP’s initial criticality assessments of the NFA store and SFA Pools (Ref. 142) considered several standard fault conditions, based on US standards, namely:

- NFA Store:
  - Ingress of water having densities over the range 0 gcm^{-3} to 1.0 gcm^{-3}. These calculations simulate flooding of the NFA store and ingress of water mists (e.g. from firefighting).
  - FA dropped on to the store during loading or unloading.
- SFA Pool:
  - Loss or dilution of soluble boron in the Pool water.
  - Fuel misloading with dropped FAs, parallel to the FAs within the Pool to simulate the effect of a loading fault.

490. The RP did not examine the effects of fuel over-enrichment of the fuel because this was judged not to be credible. I agree that the likelihood would be extremely low because of the large number of QA failures that would need to occur during FA manufacturing. FA manufacturers go to great lengths to ensure that their products are within-specification.

491. In the initial submission, I noticed that the following fault conditions (commonly considered in FA transport ((Ref. 163) (Ref. 164) (Ref. 165))) had not been addressed:

- An accumulation of fuel fragments in the SFA Pool:
  - Arising from debris generated by dropped SFAs (spent fuel can be brittle), building-up somewhere in the Pool. I consider a criticality from this scenario to be unlikely but could not rule it out, given the life of the SFA Pool (potentially 60 years or more), the likely number of fuel movements and the significant drop heights.
  - In theory, water currents within the Pool (e.g. from refilling or moving FAs) could move the debris to a position where it could form a single unit interacting with nearby FAs in the Pool, increasing overall reactivity and undermining criticality safety.
  - A full safety justification will require probabilistic arguments, but I sought assurance that the criticality safety of the SFA Pool could not be undermined by small amounts of debris. In my opinion an assessment of locations (i.e. within the FAs and especially outside of the storage envelope) was needed to fully explore this topic. After some discussion to fully explain my concerns, the RP agreed to carry out the extra work. The results are discussed below.
- Change in the geometry of an FA:
  - Change in geometry of an FA as a result of an impact and subsequent storage in the Pool. An energetic impact could alter the geometry of a FA, causing a change in the spacing between fuel pins. This could be important because the neutron multiplication produced by a FA is highly dependent on the inter fuel rod spacing. After some discussion to fully explain my concerns, the RP agreed to carry out the extra work. The results are discussed below.
- Fuel misloading/Dropped FA in the SFA Pool:
  - The exact arrangement was not initially clear from the report and did not appear to consider fuel lying across the Pool.

492. I engaged with the RP to explain the shortfalls over a number technical radiological protection meetings (Ref. 146) (Ref. 148), raising RQs where required, RQ-UKHPR1000-1077 and RQ-UKHPR1000-1381 (Ref. 20) and giving presentations to explain ONR's expectations (Ref. 147). The RP responded promptly undertaking calculations and providing justifications to satisfy my concerns. The discussions and outcomes are summarised below.

4.9.1.5 Sensitivity Analyses

493. From the outset of the project, the RP were clearly cognisant of the fact that uncertainty is present in all engineering/manufacturing parameters and therefore in the values used in criticality models, which could adversely affect criticality justifications.

494. The RP assessed uncertainties in parameters by including value ranges as sampling distributions in the Monte-Carlo analysis, a technique sometimes referred to as the "Total Monte Carlo Method" (see 'Total Monte-Carlo Method Applied to The Assessment Of Uncertainties In A Reactivity-Initiated Accident' (Ref. 166) for example). This allowed the RP to compute an error/uncertainty bound and increase all K_{eff} values by 0.02 (at the 95/95 Confidence Level), to allow for their effects.

495. Independently, ONR instructed a TSC to undertake specific calculations to gain assurance that criticality safety would not be compromised by credible variations in important model parameters (Ref. 167). Specifically, credible variations in fuel density,

separations between racks & walls, different types of Zircaloy fuel cladding, and specifications of stainless steel were examined. No significant erosion of the criticality safety margins was found.

496. The TSC's calculations are reported (Ref. 167) and confirmed that the RP's criticality justifications were not unduly sensitive to material and dimensional changes.

4.9.1.6 The Results of the RP & TSC Criticality Calculations

497. The RP's preliminary results (Ref. 142) are shown in Table 9. All Ks are less than 0.95.

Table 9: Initial RP Results

Fuel location	Model/Conditions	The RP JMCT Results	
		K_{eff}	K
1. NFA Store	1. Normal condition – Dry	0.45867	0.48367
	2. Flooded with water at 1 gcm^{-3}. (All water densities in the range: 0 - 1 gcm^{-3} were assessed and produced lower values of K_{eff} and K.)	0.90584	0.93084
	3. Dropped FA	0.56790	0.59290
2. SFA Pool	1. Normal Conditions, with 2000 ppm boron to water	No result provided. The RP argued that this was covered by Model 2.2.	
	2. Pool water dilution - no Boron in Pool water	0.90836	0.93336

K_{eff} = neutron multiplication factor as calculated by JMCT code.

$K = K_{eff} + 0.025$ to account for all potential errors and uncertainties.

Note all results are less than 0.95.

498. Initially, the TSC was unable to fully replicate the RP's results and several discussions and data exchanges were held to further understanding and resolve the differences. The values of K_{eff} obtained by the TSC were similar, but statistically significantly different from the RP's values, giving rise to a concern of a possible misunderstanding of the RP's models. The discussions are recorded in technical minutes (Ref. 146) and 'The Supplement Analysis on Criticality Safety' (Ref. 148). On each occasion, the RP responded very positively to fully resolve the issues.

499. At the end of the discussions, the TSC was able to independently (using different criticality code, nuclear data and computers) model the RP's NFA store and SFA Pool using nearly identical design data (dimensions, layouts and material specifications). Some items of data remained uncertain but were judged not to be important. The results provided ONR with confidence that the RP's calculations were sufficiently accurate.

500. The TSC's calculations are recorded in 'MCNP Verification of JMCT Criticality Calculations for Fuel Storage' (Ref. 168). The final RP results are recorded in 'Criticality Analysis of Fuel Storage' (Ref. 149). Table 10 and Table 11 provide a summary of the results.

Table 10: Final RP & Confirmatory TSC Results

Fuel Location **	Model/Condition	TSC MCNP			RP JMCT	Result ratio JMCT / MCNP
		K	sigma	K + 3σ	K-eff ††	
1. New Fuel Assembly Store	1. normal conditions	0.44216	0.00022	0.44282	0.45867	1.036
	2. normal conditions ENDF/B-VII.0 ‡‡	0.44171	0.00023	0.4424	0.45867	1.037
	3. flooded with full density water	0.90575	0.00035	0.9068	0.90584	0.999
	4. dropped fuel	0.54739	0.00026	0.54817	0.5679	1.036
2. Spent Fuel Assembly Pool	1. normal conditions 2000 ppm borated water	0.73938	0.00034	0.7404	0.73968	0.999
	2. Total loss of soluble Boron (otherwise normal)	0.90915	0.00035	0.9102	0.90836	0.998

Table 11: Fuel Debris: RP & TSC Results

Mass (kg) of fuel required for K + 3σ = 0.95 in the Spent Fuel Assembly Pool			
Model Condition	**TSC MCNP calculation**	**RP JMCT Calculation**	**Ratio (MCNP/JMCT)**
1. Fuel debris - optimally moderated spheres in borated water.	338	253	1.33
2. Fuel debris - spheres in contact with each other (under moderated) in borated water.	541	650 -720 [a]	0.83 - 0.75
Note [a]: The higher value is the estimate with no allowance (=0.025 – see Section 4.2.1.2) made for errors and uncertainties.			

501. With respect to the fault conditions described in Para. 491 (changes in the geometry of an FA dropped into the SFA Pool and a dropped FA across the NFA Store), I held discussions with the RP to explain my concerns.

- The RP explained that they had modelled a dropped FA dropped across the NFA Store, landing in both vertical and horizontal orientations, and the most reactive case is reported in Table 10 (see calculation 1.4). It can be seen that result (K=0.5679) satisfies the CSC.

** *Calculations have unique identifiers, eg 1.3 refers the NFA Store under flooded conditions.*
†† The RP K-eff in Table 2 do not contain allowances for error & uncertainty. The RP K-eff values must be increased by 0.025 for comparison with the criticality safety criteria.
‡‡ *The TSC repeated calculation 1.2 with the nuclear data set used in the JMCT calculations. A difference in k was obtained (= 0.44171 - 0.44216 = 0.00045) which is equal to the combined sigma 0.00045 (=0.00022 + 0.00023). This is not statistically significant indicating that there are no important differences between the nuclear data sets used by the TSC and the RP which would explain differences in results.*

- With regard to changes in the geometry of an FA dropped into the SFA Pool, the RP undertook additional studies – these are reported in the Criticality Analysis of Fuel Storage (Ref. 149). The RP constructed a model in which the pitch (inter-rod spacing) of the fuel rods was increased to the maximum allowed by the assembly structure along the entire length of each rod. Intermediate pitches were also considered. This is a highly conservative approach because in an impact, geometrical changes would only be expected in the end-region affected by the impact.
- The results of the RP's calculations show that an increase in the pitch would result in an increase in K, but even at the maximum this would be insufficient to breach the CSC in a SFA Pool with water containing the normal concentration of 2000 ppm boron. (The TSC was not asked to confirm these results because by this stage I had developed confidence in the RP's competence.)

502. In summary:

- All Ks are less than 0.95 and satisfy the CSC.
- In terms of magnitude, the biggest differences between the TSC and RP results (represented as the ratio of computed neutron multiplication factors, i.e. K_{JMCT}/ K_{MCNP}) is less than 4% and the smallest differences are negligible (~ 0.001).
- The largest differences in the Ks are observed at the lower reactivities (K ~0.4-0.6), which are the least important for safety because the models/conditions have the biggest safety margins.
- At the larger neutron multiplication factors (K ~0.9), the agreement between the RP and TSC models is very good. (For example: compare calculation 2.2: 0.90915 v 0.90836 – the difference is 1.8 x the pooled standard deviation, showing little, if any, statistical significance).

503. With regard to Table 11, which shows the results of calculations estimating the quantity of fuel debris needed to accumulate to cause a criticality hazard in the SFA Pool, the agreement is less good; roughly the differences are about 25 – 30% between the RP (JMCT) and TSC (MCNP) estimations.

504. Attempts were made to resolve the differences but without success [§§]. Given that these calculations represent very low probability fault scenarios, and the quantities are far above those that could inadvertently occur in the SFA Pool, I considered that it was not proportionate to pursue this line of assessment (see Provision #3 of 'Statutory Guidance: Regulators' Code' (Ref. 169)) - which requires regulators to be proportionate in their assessments) - and no further work was undertaken during Step 4 of GDA.

505. Despite the unresolved differences, the results of the calculations are very useful because they show that small quantities of uranium fuel debris cannot cause a criticality hazard in the SFA Pool. Both sets of calculations are highly pessimistic, being based on unrealistically conservative assumptions of unirradiated fuel, and idealised particle size, separation and location.

506. To put the numbers in Table 11 into context: the mass of uranium fuel in a FA will weigh several hundred kg; these accumulations would require the complete destruction of a significant part of an entire fuel assembly. In reality, considerably more fuel debris would need to accumulate to cause a criticality because the particles, separations and positions would be far from optimal; additionally, a criticality event would require a K = 1 rather than 0.95.

[§§] *The disparity may be due to differences in nuclear data, approximations within the codes, unrevealed modelling approximations or a combination of these, or even, perhaps, modelling error. However, the difference appear high when reading reported studies on code performance – see (Ref. 177) (Ref. 178)*

4.9.2 Strengths

507. In my judgement, the RP has presented a robust criticality safety case for the generic designs of the NFA store and SFA Pool:

- It has largely identified and followed RGP. In those areas where there were potential gaps to RGP, either the gaps were insignificant (V&V) or the RP readily rectified them (fault conditions).
- The generic designs show significant criticality safety margins under both normal and credible fault conditions.
- Despite the mismatch between the RP and TSC debris calculations, the RP has provided confidence that risk of criticality arising from impacts and debris accumulation in the SFA Pool is extremely low.
- I judge that the criticality risk inherent in the generic design of the NFA store and SFA Pool is ALARP, as there is little that could practically be done to further reduce the criticality risk.

4.9.3 Outcomes

508. The RP's criticality justification for the NFA store and SFA Pool was consistent with ONR's expectations. (Despite the mismatch between the results of the RP's and TSC's estimation of critical quantities, I consider that it would not be proportionate to raise any criticality-related AFs or minor shortfalls).

4.9.4 Conclusion

509. Based on the outcome of my assessment of the criticality safety of the NFA Store and SFA Pool, I have concluded that these generic designs are adequately safe. Other aspects of fuel storage (i.e. transfers into and out of the NFA store and SFA Pool and operations with damaged fuels) were not assessed.

510. From the scope of my criticality assessment of the NFA Store and SFA Pool, I have concluded that overall the approach undertaken by the RP is acceptable and fulfils the claim stipulated with the PCSR Chapter 22 (Ref. 3).

4.10 Demonstration that Relevant Risks Have Been Reduced to ALARP

4.10.1 Assessment

511. As part of the Step 4 of GDA Radiological Protection assessment plan (Ref. 5), I assessed several aspects relating to ALARP. These were:

- Evidence was sought from the RP to demonstrate that a systematic approach to option identification had been employed as part of the ALARP process for radiological protection.
- Evidence was sought from the RP on the ALARP demonstration for source term, including application of a numerical scoring system for optioneering. This applied particularly to the minimisation of cobalt and silver in the primary circuit.
- The RP has refined the occupational exposure dose estimate to reflect any changes in design, specification or proposed operational controls that affect source term or any other factors that impact occupational exposure.
- The RP has assessed the doses to workers employed in post-accident response and demonstrated that these are ALARP and compliant with legislative requirements.
- I engaged a TSC to support my assessment of the RP's approach to radiation shielding design, and to determine that the use of radiation shielding reduces occupational and public exposures SFAIRP.

512. The requirement for risks to be reduced SFAIRP arises from the Health and Safety at Work etc. Act 1974 (Ref. 133).

513. Particularly for radiological protection, ALARP is covered within IRR17 (Ref. 10) for restriction of exposure, whereby the RP must undertake necessary steps to restrict SFAIRP exposure to workers and other persons from ionising radiation.

514. TAG NS-TAST-GD-005 (Ref. 4) provides further information to ONR inspectors to help judge whether a licensee / nuclear new build has met the requirement to reduce risks to ALARP. As the generic UK HPR1000 design will be a nuclear new build it is expected the following four areas will be covered (Ref. 4):

- Clear conclusion that there are no further reasonable practicable improvements that could be implemented, and therefore the risk has been reduced to ALARP.
- The RP must set out the standards and codes used and justify them to the extent that ONR 'deem' them RGP when viewed against the SAPs.
- Reviewing of optioneering taking into account evolution of the design from its forerunners, what improvements are feasible to the design as well as providing an appropriate argument when no further work can be undertaken (not reasonably practicable).
- Risk assessments used to identify potential engineering and/or operational improvements as well as confirming numerical levels of safety.

515. The RP has a high-level overview of the ALARP methodology for the generic UK HPR1000 design within PCSR Chapter 33, however for Radiological Protection I assessed the 'ALARP demonstration Report of PCSR Chapter 22' (Ref. 170).

516. As noted in previous sections of this report I assessed the ALARP assessment for each of the Radiological Protection topic areas, this section will provide a general overview of the RP's approach for the generic UK HPR1000 safety case.

4.10.1.1 ALARP methodology

517. Within the ALARP demonstration report for PCSR Chapter 22 (Ref. 170), it provides a breakdown of the ALARP demonstration for radiological protection for the generic UK HPR1000 design.

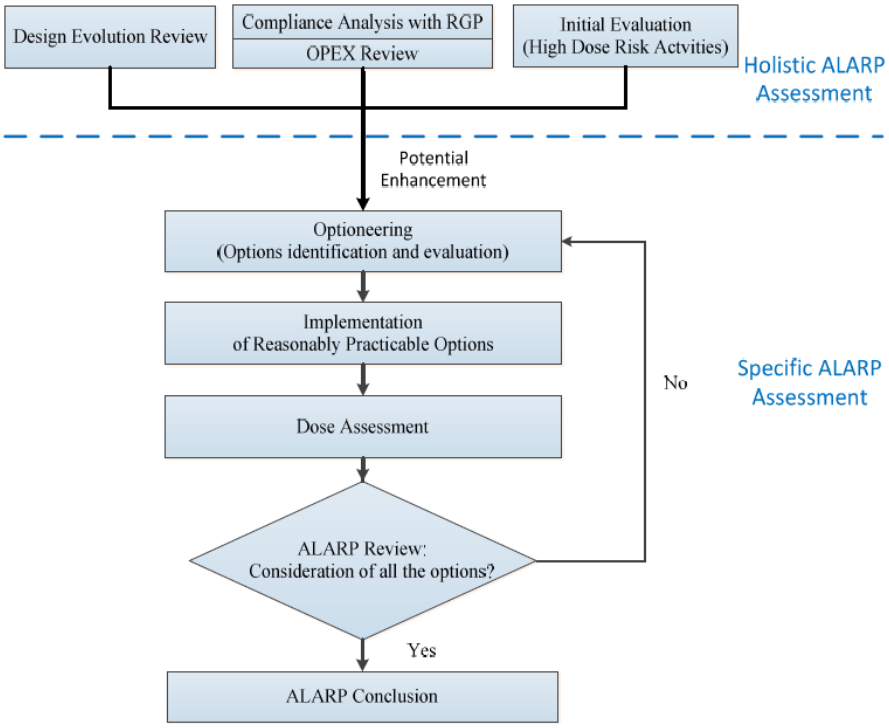

Figure 10: Process of ALARP demonstration for radiological protection (Ref. 170)

518. The approach used by the RP and presented in figure 10 aligns with the approach discussed within PCSR Chapter 33.

519. From my review of the ALARP assessment for each of the radiological protection technical areas, the majority follow the above principle when determining if the approach is ALARP. When there are differences in demonstrating ALARP, the RP has provided valid reasons. In my judgement the RP ALARP methodology is adequate and follows current UK standards and guidance.

4.10.1.2 Evidence of the Application of the RPs ALARP Process

520. Within the ALARP demonstration report (Ref. 170) references are provided of documents linked to the safety case. As can be seen in the preceding sections of this report, I sampled these documents, and found the following aspects supporting the RP's application of ALARP:

- The RP has used appropriate OPEX (from previous reactor design such as CPR1000 and M310 as well as international OPEX) to improve the generic UK HPR1000 design from a holistic ALARP perspective.
- Throughout GDA the RP has modified the generic UK HPR1000 design through DR1.0 to DR2.2. During each design review the worker dose has been further optimised.
- The RP has used appropriate ALARP methodology tools (such as ERICPPE) to reduce doses SFAIRP.
- Regarding dose to workers, the RP has demonstrated that it has reduced dose using the above ALARP methodology for high dose activities, noting that certain aspects relating to low dose activities will require further assessment which will be undertaken at the site-specific stage.
- Regarding dose to the public, the RP has demonstrated dose reduction using the ALARP methodology for the main nuclear island, noting that further substantiation relating to the waste facilities (BQF and BQZ) will be undertaken at the site-specific stage.

521. In some cases, the RP did not provide enough evidence to demonstrate risk had been reduced SFAIRP. Where this has occurred, I have raised an AF or a shortfall.

522. Overall, it is my judgement that the RP has provided appropriate evidence to demonstrate that it has applied its ALARP process to the development of the generic UK HPR1000 design.

4.10.2 Strengths

523. From the above assessment on the demonstration that relevant risks have been reduced to ALARP, I have noted the following strengths from the generic UK HPR1000 safety case:

- The ALARP methodology chosen by the RP is adequate and in line with current UK standards and guidance.
- The RP in the majority of cases has provided evidence of the ALARP methodology being used effectively.

4.10.3 Outcomes

524. The RP has provided appropriate demonstration that an ALARP approach has been undertaken from a radiological protection perspective.

525. I have raised six AFs and a number of shortfalls which relate specifically to the demonstration of ALARP. These are refereed to within previous sections.

4.10.4 Conclusion

526. Based on the outcome of my assessment the RP has demonstrated that relevant risks associated with the generic UK HPR1000 design have been reduced to ALARP. I have concluded that overall the approach undertaken by the RP is reasonable.

4.11 Consolidated Safety Case

4.11.1 Assessment

527. The focus of my assessment of the PCSR has been PCSR Chapter 22 Revision 1, however, later in Step 4 of GDA the RP submitted an updated Revision 2 of PCSR Chapter 22 (Ref. 3). I have sampled this to check consistency with the basis of my assessment and to form a view on the quality of the document.

528. For any safety case there has to be traceability and a golden thread between all tiers of documents. As stated within Section 3, the RP had four tiers of safety case documentation.

529. Through my assessment of the generic UK HPR1000 safety case I have reviewed numerous documents and on occasion the documents have not referenced the latest revision. Where these instances have occurred, I raised these with the RP and they were resolved appropriately.

530. As part of the consolidated safety case assessment for PCSR Chapter 22, I sampled a number of references within PCSR Chapter 22 for traceability (Ref. 3). From this sampling, the references to other sections of the PCSR were found to be correct.

531. It is noted that from the assessment of radiation and contamination zoning, the topic report does not reference the latest revision of some of the supporting documents. Through discussions with the RP it has been noted that the references will be updated post Step 4 of GDA.

532. Within the PCSR Chapter 22 (Ref. 3) it states that documentation will require updating to consider design modification to the generic UK HPR1000 design from DR2.2 to DR3.0. These changes could affect the collective worker dose as well as values for NT 1 and 2 hence additional assessment will have to be undertaken during the site-specific stage.

533. The documents that were revised to consider the design improvements for DR2.2 and DR3.0 were:

- 'Worker Dose Topic Report' (Revision G) (Ref. 171)
- 'ALARP Demonstration Report of PCSR Chapter 22 '(Revision F) ((Ref. 172)
- 'Evaluation of the Impacts on Collective Dose from the Design Improvements' (Revision D) (Ref. 173)

534. From reviewing the updated documents it appears there have been an additional 14 design improvements from DR2.2 to DR3.0, with only one of these improvements affecting radiological protection, which is the modification of the BFX to adopt gantry crane (Ref. 173). From the information provided this affects several topic areas of radiological protection, however the change in exposure is negligible, hence the current collective worker dose prediction remains valid.

535. The changes to the other documents (Ref. 171) (Ref. 172) are minor (updating references), hence the information presented in PCSR Chapter 22 (Ref. 3) is still valid.

536. In addition I reviewed several RQ responses where it is noted that the RP agreed to consider updating documentation in line with changes identified as appropriate. The RQs sampled were (Ref. 20):

- RQ-UKHPR1000-0850
- RQ-UKHPR1000-0999
- RQ-UKHPR1000-1516
- RQ-UKHPR1000-1609
- RQ-UKHPR1000-1611

537. From reviewing the above RQ responses against the latest submissions, I judge the RP has updated the UK HPR1000 safety case to be in line with aspects identified within the RQ responses.

538. From the review of documents and RQ responses, it is evident that the RP has consolidated information and there is clear traceability with all documents and where there are differences, this has been explained within PCSR Chapter 22 (Ref. 3).

4.11.2 Strengths

539. From the above assessment on the consolidation of PCSR Chapter 22 (Ref. 3), I have noted the following strengths from the generic UK HPR1000 safety case:

- The RP has provided appropriate traceability of documents from the PCSR Chapter 22.
- The RP has incorporated RQ responses into the latest submissions in relation to PCSR Chapter 22.
- There is a clear golden thread within all sections of the PCSR Chapter 22.

4.11.3 Outcomes

540. The RP has provided appropriate evidence to demonstrate consolidation of PCSR Chapter 22 (Ref. 3).

541. The updated documents do not change the outcome of the RP safety case for PCSR Chapter 22 (Ref. 3).

4.11.4 Conclusion

542. Based on the outcome of my assessment of consolidation of PCSR Chapter 22 (Ref. 3), I have concluded the overall approach undertaken by the RP is acceptable.

4.12 Comparison with Defined Standards, Guidance and Relevant Good Practice

543. During my assessment I have compared the RP's safety case, its claims, arguments and evidence provided against relevant legislation, standards, guidance and RGP as detailed in Section 2 and throughout the relevant sections of this report.

544. I have assessed the generic UK HPR1000 safety case against the define standards in IRR17 (Ref. 10). The RP provided the document 'Consistency Analysis of UK HPR1000 Radiological Protection Design with IRR17' (Ref. 174) detailing how the safety case aspects comply with IRR17.

545. The document (Ref. 174) provided traceability of how the generic UK HPR1000 design is consistent with the requirements of IRR17 for several of the regulations. It should be noted that as the generic UK HPR1000 design is in the design phase, some aspects of IRR17 are not relevant at Step 4 of GDA. Table 12 below provides a summary of where the RP has provided evidence of how the generic UK HPR1000 safety case will be compliant with relevant IRR17 regulations.

Table 12: Summary table of RP compliance with relevant regulation IRR17 (Ref. 10) requirements

IRR 17 Reg	Main Evidence provided	Section of the report
Regulation 8 – Radiological Risk assessment	The regulation 8 requirement for radiation risk assessment is addressed in areas I assessed. This is principally in Sections 4.3 Radiation Shielding, 4.4 Worker Dose Assessment, 4.6 Radiation and Contamination Zoning and 4.9 Criticality Safety. Radiation risk assessment methodology has been used by the RP to reduce risks in the generic UK HPR1000 design.	4.2 to 4.10
Regulation 9 – Restriction of exposure	The regulation 9 requirement for restriction of exposure is addressed in all the areas I assessed. The RP has provided details of how the hierarchy of control measures has been implemented to reduce doses SFAIRP in the generic UK HPR1000 design.	4.2 to 4.10

IRR 17 Reg	Main Evidence provided	Section of the report
Regulation 10 – Personal protective equipment	The regulation 10 requirement for personal protective equipment is addressed in Sections 4.4 Worker Dose Assessment 4.6 Radiation and Contamination Zoning. The RP has provided examples of the types of PPE available to restrict worker doses and for use in designated areas of the generic UK HRP1000 design. The actual selection and use of personal protective equipment will be addressed by the licensee at the site-specific stage.	4.4 and 4.6
Regulation 12 – Dose Limitation	The regulation 12 requirement for dose limitation is addressed within Sections 4.4 Worker Dose and 4.5 Public Dose Assessment. The RP has provided substantiation that worker and public dose limits will not be exceeded for the generic UK HPR1000 design.	4.4 and 4.5
Regulation 13 – Contingency Plans	The regulation 13 requirement for contingency plans is addressed within Section 4.8 Post Accident Accessibility and Emergency Arrangements. The RP has provided an outline of what will be required for the generic UKHPR1000 design with regard to contingency plans and emergency arrangements. These will be developed by the licensee at the site-specific stage.	4.8
Regulation 17 – Designation of Controlled or Supervised areas	The regulation 17 requirement for designation of controlled and supervised areas is addressed within Sections 4.3 Radiation Shielding and 4.6 Radiation and Contamination Zoning. The RP has provided an example of an approach to zoning for the generic UK HPR1000 design and the licensee will determine its actual approach at the site-specific stage.	4.3 and 4.6
Regulation 19 – Additional requirements for designated areas	The regulation 19 requirement for additional requirements for designated areas is addressed within Section 4.6 Radiation and Contamination Zoning. The RP has provided an example of an approach for the generic UK HPR1000 design and the licensee will determine its actual approach at the site-specific stage.	4.6

IRR 17 Reg	Main Evidence provided	Section of the report
Regulation 20 – Monitoring of designated areas	The regulation 20 requirement for monitoring of designated areas is addressed within Section 4.7 Radiation and Contamination Monitoring. The RP has provided an example of an approach to monitoring for generic UK HPR1000 design purposes and the licensee will determine its actual approach at the site-specific stage.	4.7

546. As can be seen from table 12, the RP has provided evidence for compliance with relevant regulations of IRR17 (Ref. 10), which cover a range of technical areas. Hence, I judge the RP has demonstrated that the generic UK HPR1000 design can be operated in compliance with IRR17 (Ref. 10).

547. It should be noted that in some cases the RP has not used the most up to date guidance. Where this has occurred, I judged the gap to be minimal and have raised a minor shortfall where appropriate.

548. Overall, the radiological protection aspects of the generic UK HPR1000 safety case have been found to be consistent with the relevant legislation, standards, guidance and RGP.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

549. This report presents the findings of my Radiological Protection and Criticality assessment of the generic UK HPR1000 design as part of the Step 4 of GDA process.

550. Based on my assessment, undertaken on a sampling basis, I have concluded the following:

- The RP has provided appropriate arguments and evidence to corroborate the claims made within the PCSR Chapter 22, Radiological Protection.
- The RP has provided appropriate arguments and evidence to corroborate the claims made within the PCSR Chapter 5, Reactor Core, from a criticality safety for fuel storage perspective.
- The RP has provided appropriate arguments and evidence to corroborate the claims made within the PCSR Chapter 32, Emergency Arrangements.
- From the assessment the RP has met the expectations of relevant guidance, SAPs and TAGs and RGP identified within the report; and
- From the assessment, nine AFs have been identified. It is my opinion that these can be resolved by the licensee during site-specific stages.

551. Overall, based on my sample assessment of the safety case for the generic UK HPR1000 design undertaken in accordance with ONR's procedures, I am satisfied that the case presented within the PCSR and supporting documentation is adequate. On this basis, I am content that a DAC should be granted for the generic UK HPR1000 design from a Radiological Protection and Criticality perspective.

5.2 Recommendations

552. Based upon my assessment detailed in this report, I recommend that:

- **Recommendation 1**: From a Radiological Protection and Criticality perspective, ONR should grant a DAC for the generic UK HPR1000 design.

- **Recommendation 2**: The nine AFs identified in this report should be resolved by the licensee for a site-specific application of the generic UK HPR1000 design.

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Annex 1

Relevant Safety Assessment Principles Considered During the Assessment

SAP Number	SAP Title	Notes
FP.3	Optimisation of Protection	Protection must be optimized to provide the highest level of safety that is reasonably practicable
FP.4	Safety Assessment	The duty holder must demonstrate effective understanding of the hazards and their control for a nuclear 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 protected against radiation risks.
RP.1	Normal Operation (Planned Exposure Situations)	Adequate protection against exposure to radiation and radioactive substances in normal operation should be provided in those parts of the facility to which access needs to be gained
RP.2	Fault and accident conditions (Emergency Exposure Situations)	Adequate protection against exposure to radiation and radioactive contamination in accident conditions, should they occur, should be provided in those parts of the facility to which access needs to be gained. 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 exposure and prevent the spread of radioactive substances

SAP Number	SAP Title	Notes
RP.4	Contaminated areas	Appropriate provisions for protecting persons entering and working in contaminated areas should be provided.
RP.5	Decontamination	Suitable and sufficient decontamination provisions for the 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 conditions.
RP.7	Hierarchy of control measures	The dutyholder should establish a hierarchy of control measures to optimise protection in accordance with IRR17.
NT.1	Normal Operations - Any Person on Site Normal Operations – Any Group on Site Normal Operations – Any Person off the site	A safety case should be assessed against numerical targets and legal limits for normal operation, design basis faults, and radiological accident risks to people on and off the site.
NT. 2	Time at Risk	There should be sufficient control of radiological hazards at all times.
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	A nuclear facility should be so designed and operated that defence in depth against potentially significant faults or failures is achieved by the provision of several levels of protection.
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).

SAP Number	SAP Title	Notes
ECR. 1	Criticality Safety – Safety measures	Wherever a significant amount of fissile material may be present, there should be safety measures to protect against unplanned criticality.
ECR. 2	Criticality Safety – Double Contingency approach	Criticality safety cases should employ the double contingency approach.
FA. 2	Fault analysis – General identification of initiating faults	Fault analysis should identify all initiating faults having the potential to lead to any person receiving a significant dose of radiation, or to a significant quantity of radioactive material escaping from its designated place of residence or confinement.

Annex 2

Assessment Findings

Number	Assessment Finding	Report Section
AF-UKHPR1000-0096	The licensee shall demonstrate how the deposited corrosion product source term is applied and used within the safety case. Corrosion products are a significant source of operational radiation exposure and the licensee shall optimise this source term to reduce worker dose so far as is reasonably practicable.	4.2.1.3
AF-UKHPR1000-0097	The licensee shall ensure that the operating experience used as evidence for the source term for normal operations is robustly underpinned and documented within the safety case.	4.2.1.3
AF-UKHPR1000-0098	The licensee shall demonstrate that the source terms used in radiation shielding assessments are justified and robust. This should address the shortfalls identified during Step 4 of GDA, including but not limited to, ensuring that radionuclide source terms for shielding assessments are correctly decayed and incorporate equilibrium daughter products, where appropriate.	4.3.1.1
AF-UKHPR1000-0099	The licensee shall, during detailed design, demonstrate that all measures have been implemented to reduce doses to workers from radiation shielding penetrations, so far as is reasonably practicable, and ensure the associated effects on radiation zoning are minimised.	4.3.1.4
AF-UKHPR1000-0100	The licensee shall review and complete its radiation shielding assessments, extending these to the totality of the detailed design and addressing the shortfalls identified during Step 4 of GDA. This should include, but is not limited to, addressing shortfalls regarding modelling assumptions supporting radiation shielding calculations and the statistical accuracy of results.	4.3.1.4

Number	Assessment Finding	Report Section
AF-UKHPR1000-0101	The licensee shall during detailed design, demonstrate that the design, operation and procedures for the removal of In Core Instrument Assemblies are optimised to reduce worker dose so far as is reasonably practicable.	4.4.1.2
AF-UKHPR1000-0102	The licensee shall during detailed design, complete the assessment of, and demonstrate that, collective worker dose for the UK HPR1000 design in totality, is reduced so far as is reasonably practicable. This should include the 40% of collective dose arising from routine activities which require detailed design and operational strategies to be addressed adequately.	4.4.1.4
AF-UKHPR1000-0103	The licensee shall demonstrate that the doses received by a worker to the lens of the eye have been reduced so far as is reasonably practicable. Given recent legislative changes in dose limits to the lens of the eye, this should include addressing the shortfalls identified during Step 4 of GDA, including but not limited to, identification of reasonably practicable measures to reduce doses to the lens of the eye and the use of international operational experience.	4.4.5
AF-UKHPR1000-0105	The licensee shall during detailed design, demonstrate local intervention worker internal radiation exposures are reduced so far as is reasonably practicable. Justification should be provided that the respiratory protection factor of 1000, used to calculate internal radiation exposure, is appropriate. This is required where the safety case identifies the necessity for respiratory protective equipment to be worn during/following accidents, which includes: • design basis accidents; • design extension condition A; and • severe accidents	4.8.1.1