Office for Nuclear Regulation
An agency of HSE

Generic Design Assessment – New Civil Reactor Build

Step 4 Radiological Protection Assessment of the Westinghouse AP1000® Reactor

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PREFACE

The Office for Nuclear Regulation (ONR) was created on 1st April 2011 as an Agency of the Health and Safety Executive (HSE). It was formed from HSE's Nuclear Directorate (ND) and has the same role. Any references in this document to the Nuclear Directorate (ND) or the Nuclear Installations Inspectorate (NII) should be taken as references to ONR.

The assessments supporting this report, undertaken as part of our Generic Design Assessment (GDA) process, and the submissions made by Westinghouse relating to the AP1000® reactor design, were established prior to the events at Fukushima, Japan. Therefore, this report makes no reference to Fukushima in any of its findings or conclusions. However, ONR has raised a GDA Issue which requires Westinghouse to demonstrate how they will be taking account of the lessons learnt from the events at Fukushima, including those lessons and recommendations that are identified in the ONR Chief Inspector's interim and final reports. The details of this GDA Issue can be found on the Joint Regulators' new build website www.hse.gov.uk/newreactors and in ONR's Step 4 Cross-cutting Topics Assessment of the Westinghouse AP1000® reactor.

EXECUTIVE SUMMARY

This report presents the findings of the Radiological Protection assessment of the AP1000 reactor undertaken as part of Step 4 of the Health and Safety Executive's Generic Design Assessment. The assessment has been carried out on the December 2009 Pre-construction Safety Report and supporting documentation submitted by Westinghouse during Step 4.

This assessment has followed a step-wise-approach in a claims-argument-evidence hierarchy. In Step 3 the claims made by Westinghouse were examined, and most of the arguments that underpin those claims were examined.

The scope of the Step 4 assessment was to review the safety aspects of the AP1000 reactor in greater detail, by examining the evidence, supporting arguments and claims made in the safety documentation, building on the assessments already carried out for Step 3, and to make a judgement on the adequacy of the Radiological Protection information contained within the Pre-construction Safety Report and supporting documentation.

It is seldom possible, or necessary, to assess a safety case in its entirety, therefore sampling is used to limit the areas scrutinised, and to improve the overall efficiency of the assessment process. Sampling is done in a focused, targeted and structured manner with a view to revealing any topic-specific, or generic, weaknesses in the safety case. To identify the sampling for radiological protection assessment, an assessment plan for Step 4 was set-out in advance.

My safety assessment within this topic includes external radiation hazards associated with direct radiation from structures, systems and components; in addition to internal radiation hazards resulting from the generation of surface and airborne contamination. I have considered the adequacy of features incorporated within the design which are intended to reduce the exposure of workers and the public under normal conditions.

My assessment has focussed on areas relevant to normal operation:

- Radiation sources.
- Designated areas (radiological classification of areas / radiological zoning).
- Shielding.
- Contaminated Areas.
- Ventilation.
- Radiological instrumentation.
- Decontamination.
- Optimisation for work activities (including fuel route).
- Waste handling and decommissioning.
- Public exposure from direct shine (direct radiation originating from within the site boundary).

My assessment has also considered areas relevant to accident conditions:

- Persons on-site.
- Intervention personnel.

A number of items have been agreed with Westinghouse as being outside the scope of the Generic Design Assessment process and hence have not been included in my assessment, such as the assessment of doses associated with mid-loop working, which will be assessed at the site specific phase should the licensee wish to undertake this practice.

From my assessment, I have concluded that:

- The design of the radiation protection aspects of the AP1000 is broadly in line with my expectations in relation to current national and international standards, guidance and relevant good practice.
- Westinghouse has demonstrated that it has made systematic improvements to the radiological protection aspects of the design throughout the design process, including the adequate development of shielding structures which have been adapted to the specific radiological conditions associated with an AP1000 reactor.
- Overall, I believe that, in the majority of areas, the AP1000 Pre-construction Safety Report has been informed by a thorough and robust analysis of the threats posed by radiological hazards coupled with a clear philosophy and logic associated with design.

In some areas there has been a lack of detailed information which has limited the extent of my assessment. As a result The (HSE) Nuclear Directorate will need additional information to underpin my conclusions and these are identified as Assessment Findings to be carried forward as normal regulatory business. These are listed in Annex 1. Assessment findings include concerns regarding the design of health physics facilities, where the amount of space allocated to facilities such as laboratories and changerooms is judged to be insufficient. As a result, the licensee will be required to improve the design and layout of the site specific health physics facilities and provide a justification that the new design reduces worker doses, and reduces the likelihood and severity of reasonably foreseeable radiological accidents, so far as is reasonably practicable.

One observation identified within this report is of particular significance and will require resolution before The Health and Safety Executive would agree to the commencement of nuclear safety-related construction of an AP1000 reactor in the UK. This is identified in this report as a Generic Design Assessment Issue and is detailed within Annex 2. In summary this relates to:

- Ensuring that criticality control of the Spent Fuel Pool is maintained by geometrical control and fixed poisons alone.

Overall, based on the samples undertaken in accordance with The (HSE) Nuclear Directorate procedures, I am broadly satisfied that the claims, arguments and evidence laid down within the Pre-construction Safety Report and supporting documentation submitted as part of the Generic Design Assessment process present an adequate safety case for the generic AP1000 reactor design. The AP1000 reactor is therefore suitable for construction in the UK, subject to satisfactory progression and resolution of the Generic Design Assessment Issue to be addressed during the forward programme for this reactor and assessment of additional information that becomes available as the Generic Design Assessment Design Reference is supplemented with additional details on a site-by-site basis.

LIST OF ABBREVIATIONS

ACOP	Approved Code Of Practice
ADS	Automatic Depressurisation System
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
ASN	Autorité de Sûreté Nucléaire (French Nuclear Safety Authority)
BMS	(Nuclear Directorate) Business Management System
BSL	Basic Safety Level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
CEDE	Committed Effective Dose Equivalent
CRDM	Control Rod Drive Mechanism
CV	Containment Vessel
CVS	Chemical and Volume Control System
DCD	Design Control Document
DCP	Design Change Proposal
DECC	Department of Energy and Climate Change
DfT	Department for Transport
EC	Eddy Current
EPR10	Environmental Permitting Regulations 2010
GDA	Generic Design Assessment
GRS	Gesellschaft für Anlagen und Reaktorsicherheit mbH
HPA-CRCE	Health Protection Agency’s Centre for Radiation, Chemical and Environmental Hazards
HSE	The Health and Safety Executive
HSWA74	Health and Safety at Work etc Act 1974, as amended
HVAC	Heating, Ventilation, and Air Conditioning
HX	Heat Exchanger
IAEA	The International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IHP	Integrated Head Package
IIS	In-core Instrumentation System
IRR99	Ionising Radiations Regulations 1999
IRWST	In-Containment Water Storage Tank

LIST OF ABBREVIATIONS

ISI	In-Service Inspection
JEM	Job Exposure Model
LLW	Low Level Waste
LNB	Lower Neutron Block
MCNP	Monte Carlo N-Particle (Shielding Code)
MDEP	Multi-national Design Evaluation Programme
MHSWR99	Management of Health and Safety at Work Regulations 1999, as amended
ND	The (HSE) Nuclear Directorate
NDA	Nuclear Decommissioning Authority
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
OCNS	Office for Civil Nuclear Security
OECD	Organisation for Economic Cooperation and Development
PCCS	Passive Containment Cooling System
PCER	Pre-construction Environment Report
PCSR	Pre-construction Safety Report
POCO	Post Operational Clean Out
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
PWR	Pressurised Water Reactor
PXS	Passive Core Cooling System
RCA	Radiologically Controlled Area
RCF	Reactor Coolant Filter
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REPPIR	Radiation (Emergency Preparedness and Public Information) Regulations 2001
RGP	Relevant Good Practice
RI	Regulatory Issue
RIA	Regulatory Issue Action
RMS	Radiation Monitoring System
RNS	Normal Residual Heat Removal System
RO	Regulatory Observation
ROA	Regulatory Observation Action

LIST OF ABBREVIATIONS

RPV	Reactor Pressure Vessel
RVCH	Reactor Vessel Closure Head
RVIS	Reactor Vessel Insulation System
SAP	Safety Assessment Principles
SFP	Spent Fuel Pool
SFS	Spent Fuel Pool Cooling System
SG	Steam Generator
SRMP	Standard Radiation Monitoring Program
SSC	Systems, Structures and Components
SSER	Safety, Security and Environmental Report
SSM	The Swedish Nuclear Safety Authority
STUK	The Finish Nuclear Safety Authority
TAG	(Nuclear Directorate) Technical Assessment Guide
TEDE	Total Effective Dose Equivalent
TQ	Technical Query
TSC	Technical Support Contractor
US NRC	Nuclear Regulatory Commission (United States of America)
VLLW	Very Low level Waste
WRS	Radioactive Waste Drain System

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1 INTRODUCTION

1 This report presents the findings of the Step 4 Radiological Protection assessment of the Westinghouse December 2009 AP1000 reactor Pre-construction Safety Report (PCSR) (Ref. 11) and supporting documentation provided by Westinghouse under the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process. Assessment was undertaken of the PCSR and the supporting evidentiary information derived from the Master Submission List (Ref. 13). The approach taken was to assess the principle submission, i.e. the PCSR, and then undertake assessment of the relevant documentation sourced from the Master Submission List on a sampling basis in accordance with the requirements of The (HSE) Nuclear Directorate (ND) Business Management System (BMS) procedure AST/001 (Ref. 2) and procedure AST/003 (Ref. 3). The Safety Assessment Principles (SAP) (Ref. 4) have been used as the basis for this assessment. Ultimately, the goal of assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety case.

2 During the assessment a number of Technical Queries (TQ) and Regulatory Observations (RO) were issued and the responses made by Westinghouse assessed. Where relevant, detailed design information from specific projects for this reactor type has been assessed to build confidence and assist in forming a view as to whether the design intent proposed within the GDA process can be realised.

3 A number of items have been agreed with Westinghouse as being outside the scope of the GDA process and hence have not been included in this assessment.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR RADIOLOGICAL PROTECTION

4 The intended assessment strategy for Step 4 for the Radiological Protection topic area was set out in an assessment plan that identified the intended scope of the assessment and the standards and criteria that would be applied. This is summarised below:

2.1 Assessment Plan

5 The Step 4 Radiological Protection Assessment Plan for the AP1000 (Ref.1) described the assessment process within ND and summarised the assessment findings in the Step 3 Radiological Protection Assessment of the Westinghouse AP1000 (Ref. 6). The Plan summarised the scope of the assessment, standards and criteria used to judge radiological protection aspects of the AP1000, interfaces with other assessment areas, liaison with other regulators, and working with technical support contractors.

2.2 Standards and Criteria

6 The key piece of legislation for nuclear facilities is the Nuclear Installations Act 1965, as amended (Ref. 25). The standards and criteria that were used to judge radiological protection in the AP1000 are legislation, SAPs (Ref. 4) and Technical Assessment Guides (TAG). The key piece of radiological protection legislation is the Ionising Radiations Regulations 1999 (IRR99) (Ref. 19). These Regulations implement the European Basic Safety Standards Directive (Ref. 26), which in turn takes into account recommendations from the International Commission on Radiological Protection (Ref. 27). Areas of particular importance to GDA Step 4 include restriction of exposure (including the hierarchy of control measures), dose limitation, designation of controlled or supervised areas, monitoring of designated areas, and duties of manufacturers. Other important pieces of legislation include the Radiation (Emergency Preparedness and Public Information) Regulations 2001 (REPPIR) (Ref. 28), Management of Health and Safety at Work Regulations 1999, as amended (MHSWR99) (Ref. 29) and Environmental Permitting Regulations 2010 (EPR10) (Ref. 30).

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

- Exposures to radiation should be justified. Justification is not regulated by HSE and is not considered in the SAPs. Justification for electrical power generation is covered by the Department for Energy and Climate Change (DECC).
- Exposures to ionising radiation should be optimised. Radiation exposures must be restricted "so far as is reasonably practicable" under IRR99, that is, doses should be "as low as reasonably practicable" (ALARP). In this report the UK term "ALARP" is taken to be synonymous with the international term "ALARA" ("as low as reasonably achievable") and with "so far as is reasonably practicable".
- Exposures to ionising radiation should be limited in that they must not exceed the statutory dose limits in IRR99 (Ref. 19). Clearly this should not be an issue for modern nuclear plant under normal operation, as is indeed the case for the AP1000 (Ref. 6).

8 Radiological protection will make a contribution to fulfilling the expectations of some of the fundamental principles in the SAPs (Ref. 4), although radiological protection, or indeed any other single topic area, could not fulfil those expectations alone. The key

fundamental principles that have some relevance to radiological protection are FP.3 to FP.8. The radiation protection principles (RP.1 to RP.6) are for normal operation, accident conditions, designated areas, contaminated areas, decontamination, and shielding, and all of these areas were covered by the assessment. This section of the SAPs on Radiation Protection (Ref. 4) also refers to IRR99 (Ref. 19), and in particular to the Approved Code of Practice (ACOP) and guidance to IRR99 on the hierarchy of control measures in regulation 8 (Ref. 22). The criticality safety principles (ECR.1 and ECR.2) are for safety measures and double contingency approach, and these areas were covered by the assessment.

9 All the numerical targets and legal limits (NT.1 Targets 1 to 9 and NT.2) are relevant to a degree. The radiological protection assessment focused on NT.1 Targets 1 to 3 regarding impacts to people during normal operation, and NT.2 regarding time of exposure of employees in high dose rate locations. The lead for design basis fault sequences and Level 3 Probabilistic Safety Assessment (PSA) was taken by ND assessors in other disciplines. The radiological protection assessment contributed to NT.1 Target 4 regarding radiological consequence assessment of design basis fault sequences and to NT.1 Targets 5 to 9 regarding radiological consequence assessment of accidents (including Level 3 PSA, which is reported in the Step 4 Probabilistic Safety Assessment of the AP1000, Ref. 31). These principles, targets and limits were assessed on a sampling basis to the extent that they could be accommodated within the GDA process. They will also need to be considered during the site specific phase.

10 IRR99 (Ref. 19) requires that, in general, the annual dose limit for workers is 20 $mSvy^{-1}$. The Basic Safety Level (BSL) as specified in the SAPs is the level of dose above which the risk of harm is intolerable and for workers who are working with ionising radiation during normal operation (NT.1 Target 1), it is the same value as the annual dose limit under IRR99 (Ref. 19), namely 20 $mSvy^{-1}$. The BSL for groups of persons working with ionising radiation during normal operation is half of that value, namely, 10 $mSvy^{-1}$ (NT.1 Target 2). The BSL for other persons on-site during normal operation (e.g. workers not working with ionising radiation, visitors) is 2 $mSvy^{-1}$ (NT.1 Target 1). The BSL for members of the public off the site during normal operation is the same as the public dose limit under IRR99 (Ref. 19), namely 1 $mSvy^{-1}$ (NT.1 Target 3).

11 The Basic Safety Objective (BSO), as specified in the SAPs, is the level below which it would not be reasonable use of ND resources to seek further reductions in radiation doses from operators. Nevertheless, the principle of ALARP still applies to operators at levels below the BSO which may drive doses down below the BSO. The BSO for workers who are working with ionising radiation during normal operation is one twentieth of the BSL / annual dose limit under IRR99 (Ref. 19), namely 1 mSv y^{-1} (NT.1 Target 1). The BSO for groups of persons working with ionising radiation during normal operation is also one twentieth of the BSL, namely, 0.5 $mSvy^{-1}$ (NT.1 Target 2). The BSL for other persons on-site during operation (e.g. workers not working with ionising radiation, visitors) is again one twentieth of the BSL, namely 0.1 $mSvy^{-1}$ (NT.1 Target 1). The BSO for members of the public off the site during normal operation is more challenging in that it is a much lower proportion (one fiftieth) of the BSL / public dose limit under IRR99 (Ref. 19), namely 0.02 $mSvy^{-1}$ (NT.1 Target 3).

12 BSLs for design bases fault sequences (NT.1 Target 4) for any people on or off the site are expressed in terms of radiation dose and are dependent on frequencies of initiating fault sequences. However, there is only one BSO for people on the site, and a different one for people off the site (also expressed in terms of radiation dose), and these are independent of frequencies of initiating fault sequences. BSLs and BSOs for accident

conditions for any people on the site or any people off the site (NT.1 Targets 5 and 6, and 7 and 8 respectively) are dependent on frequencies of accidents.

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

14 The TAGs of most relevance to the assessment are on fundamental principles (Ref. 15), demonstration of ALARP (Ref. 32), radiological protection (Ref. 16), shielding (Ref. 14), criticality safety (Ref. 33), criticality warning systems (Ref. 34), radiological analysis during normal operation (Ref. 17), and radiological analysis during fault conditions (Ref. 18).

15 The relevant fundamental principles, radiation protection principles, criticality safety principles and numerical targets and legal limits from the SAPs (Ref. 4) are summarised in Table 12, along with relevant Western European Nuclear Regulators' Association (WENRA) and International Atomic Energy Agency (IAEA) references (Refs 7, 20, 21 and 35, respectively) and TAGs (Refs 14, 15, 16, 17, 18, 32, 33, and 34). This table also indicates the contributions made by these principles, targets and limits to the Step 4 radiological protection assessment. Since the Step 4 Radiological Protection Assessment Plan (Ref. 1) was prepared, the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development has also published guidance on occupational radiological protection principles and criteria for designing new nuclear power plants (Ref. 36). This document provided useful guidance for this assessment.

16 The principal standards and criteria for judging whether ALARP has been met are the ACOP and guidance to IRR99 (Ref. 22), supplemented by additional guidance on HSE's website (including the TAGs). In addition, IRR99 (Ref. 19) require a hierarchical approach to control exposure: first, exposures should be restricted by engineered controls and design features (and in addition, by the provision and use of safety features and warning devices); secondly, by supporting systems of work; and thirdly and lastly, by the provision of personal protective equipment.

17 The principal standards and criteria for judging whether ALARP has been met for intervention personnel during accident conditions is in the Guide to REPPIR (Ref. 37), supplemented by additional guidance on HSE's website (Ref. 38). The Health Protection Agency's Centre for Radiation, Chemical and Environmental Hazards (HPA-CRCE) has published guidance on controlling doses for people on-site during radiation accidents (Ref. 39).

18 When judging against the ALARP principle, caution should be used to distinguish between dose and risk. The general duties of employers to their employees and other persons in Sections 2 and 3 respectively of the Health and Safety at Work etc Act 1974, as amended (Ref. 23), refer to risks as do the expectations in many of the SAPs (Ref. 4). However, the duties of radiation employers in IRR99 (Ref. 19) and standards in some of the SAPs (Ref. 4) refer to radiation exposures and not just to the implied risk. The hierarchy of control measures in IRR99 (Ref. 19) is also applicable here, as the Approved Code of Practice (ACOP) to regulation 8 advises radiation employers to give priority to improving engineering controls and adopting other means of restricting exposure over and above dose sharing between employees (Ref. 22). If a choice has to be made between restricting exposures to individuals or to groups of employees then priority should always be given to restricting exposures to individuals. In contrast to this, under accident conditions, the risk is determined by both the magnitude of the dose and the probability of its occurrence. For the purposes of ALARP, the risk of harm to an individual from whole-body exposure is taken to be directly proportional to that dose.

19 The ALARP principle applies to the exposure of members of the public. The regulation of public radiation exposure during normal reactor operation is shared between the Environment Agency and HSE, where IRR99 (Ref. 19) is enforced by ND on behalf of HSE, and EPR10 (Ref. 30) is enforced by the Environment Agency. IRR99 (Ref. 19) require dose constraints to restrict exposure to ionising radiation at the planning stage where it is appropriate to do so. The guidance to IRR99 (Ref. 22) advises that a constraint for a single new source should not exceed 0.3 mSv per year for members of the public. This is reinforced in the SAPs (Ref. 4) in relation to NT.1 Target 3 and advises that HSE's view is that a single source should be interpreted as a site under a single dutyholder's control, since this is an entity for which radiological protection can be optimised as a whole. However, the former Health Protection Agency's Centre for Radiation, Chemical and Environmental Hazards (HPA-CRCE) has recently recommended that the dose constraint for members of the public from new NPPs should be 0.15 mSv per year (Ref. 24).

20 The ALARP principle also applies to manufacturers, etc. Section 6 of HSWA74 (Ref. 23) places general duties on manufacturers, etc. as regards articles and substances for use at work and duties on any person who designs, manufactures, imports or supplies any article for use at work. Where that work is with ionising radiation, the duty is modified to apply to articles for use at work by IRR99, regulation 31 (Ref. 19). This requires manufacturers, etc. to apply the ALARP principle in that there is a duty to ensure that any such article is so designed and constructed as to restrict so far as is reasonably practicable, the extent to which employees and other persons are, or are likely to be, exposed to ionising radiation. Therefore, the requirement in law to keep radiation exposures ALARP applies not only to the licensee of a NPP, but also to the designer of that NPP.

2.3 Assessment Scope

21 The objective of the Step 4 assessment was to review the safety aspects of the proposed reactor designs in more detail by examining the evidence supporting arguments and claims made in the Westinghouse safety documentation, and by building on the assessment already carried out for Step 3, in order to make a judgement on the adequacy of the radiological protection aspects of the revised PCSR and supporting documentation.

22 The Step 4 assessment assessed whether occupational and public exposures to ionising radiations are ALARP during normal operation. This assessment re-visited the Step 3 assessment in light of detailed evidence submitted by Westinghouse and assessed the robustness of that evidence for potential dose uptake. The assessment focused on areas not covered in Step 3, such as occupational exposure associated with the fuel route, waste handling, shielding, ventilation, contamination control, radiological instrumentation, and decommissioning. Other assessors looked at accident risk and the radiological protection assessment contributed to the analysis of Level 3 PSA, with regard to plume dispersion modelling and dose consequences during Step 4. As already noted, the Level 3 PSA assessment is reported in the Step 4 Probabilistic Safety Assessment of the Westinghouse AP1000 (Ref. 31).

23 There are matters relevant to radiological protection that cannot be adequately assessed during GDA, as they are directly related to the operating regimes selected by future licensees. This assessment has been primarily focused on the radiological risks associated with physical design features associated with the AP1000, rather than the specific working practices where there is an inherent radiological risk, because these

practices will be subject to change based on licensee operating preferences. However, Westinghouse has submitted examples of specific working practices for some tasks in order to demonstrate that the magnitude of doses incurred by personnel, align with relevant legislation and standards and to demonstrate the effectiveness of design features, which have been incorporated within the AP1000 plant in order to restrict exposure to ionising radiations. These examples have been a useful factor in demonstrating the application of the ALARP principle.

24 The assessment was carried out in consultation with assessors in ND and the Environment Agency in other topic areas, such as PSA, deterministic safety analysis (fault studies), reactor chemistry, radioactive waste management, decommissioning, mechanical engineering, human factors, environment and control and instrumentation, as necessary.

25 A number of other topic areas in the SAPs (Ref. 4) have some relevance to radiological protection, such as safety cases, siting (not a direct issue for the GDA process), key principles, integrity of metal components and structures, layout, control of nuclear matter, control and instrumentation of safety-related systems, containment and ventilation, heat transport systems, radioactive waste management, and decommissioning. The lead for these topic areas was taken by ND assessors in other disciplines and this assessment contributed to radiological protection aspects of these topic areas as appropriate.

2.3.1 Findings from GDA Step 3

26 Much of radiological protection depends on detailed design and so some conclusions drawn at the end of Step 3 (and Step 4) have to be provisional until the design is finalised. Also, some matters may not be wholly appropriate for the GDA process and would also need to be addressed at the site specific phase by the licensee. In such cases, the design would need to be sympathetic to the potential needs of the licensee.

27 The Step 3 Assessment Report (Ref. 6) concluded that the vast majority of the claims that were assessed were appropriate and all of the arguments that were assessed were adequate (more areas were assessed for their claims than for their arguments). No Regulatory Observations (RO), Regulatory Issues (RI) or potential GDA Issues were identified.

28 The Step 3 Assessment Report (Ref. 6) concluded that Westinghouse had provided a reasonable safety analysis of radiological protection during normal reactor operation and that the majority of the claims and all of the arguments assessed for radiation doses being ALARP were adequate for GDA Step 3.

2.3.2 Additional Areas for Step 4 Radiological Protection Assessment

29 The additional areas for further assessment during Step 4 were listed in Table 3 of the Step 4 Radiological Protection Plan (Ref. 1). These assessment areas were split into those relevant to normal operation and those relevant to accident conditions.

30 The assessment areas relevant to normal operation are summarised below.

- Radiation sources.
- Designated areas (radiological classification of areas / radiological zoning).
- Shielding.

- Contaminated Areas.
- Ventilation.
- Radiological instrumentation.
- Decontamination.
- Optimisation for work activities (including fuel route).
- Waste handling and decommissioning.
- Public exposure from direct shine (direct radiation originating from within the site boundary).

31 The assessment areas during accident conditions are summarised below.

- Persons on-site.
- Intervention personnel.
- Off-site radiological consequence assessment (Level 3 PSA is reported in the Step 4 PSA Assessment of the Westinghouse AP1000 (Ref. 31)).

2.3.3 Use of Technical Support Contractors

32 A Technical Support Contractor (TSC) AMEC was engaged to assist in the topic areas of radiological protection and radioactive waste management and decommissioning during Step 3 and Step 4. The project was to report a literature review of radiological protection and radioactive waste management practices during the last ten years of normal operation of pressurised water reactors (PWR) (Ref. 80).

33 More TSCs were engaged to assist with the radiological protection assessment work during Step 4 and are summarised below.

- Nuclear Technologies (NT) undertook a detailed technical review of shielding (Ref. 40).
- NT / TÜV SÜD undertook a detailed technical review of general radiological protection and, in particular, optimisation of high dose work activities (Ref. 41).
- GRS undertook a detailed technical review of criticality control in the SFP (Ref. 42), and ND’s review of this work is included in Appendix A.
- REACT Engineering undertook a detailed technical review of decontamination and decommissioning (Ref. 43). The radiological protection aspects of these topic areas are summarised in my report. However, most of this work is reported in the Step 4 Radioactive Waste and Decommissioning Assessment of the AP1000 (Ref. 44).
- HPA-CRCE undertook a detailed technical review of plume dispersion modelling and dose consequences for Level 3 PSA and the findings of this review are incorporated into Ref. 31.

34 Whilst the TSCs undertook detailed literature and technical reviews, these reviews were under close direction and supervision by ND and the regulatory judgments on the adequacy, or otherwise, of the radiological protection aspects of the AP1000 were made exclusively by ND. The findings relating to radiological protection aspects of the literature and technical reviews by TSCs are incorporated into Section 4 of my report, as

appropriate, with the exception of the findings regarding the technical review of Level 3 PSA which are incorporated into Ref. 31.

35 Following due regulatory process, the visibility of TSC work and feedback on progress and outcomes of TSC work was provided to Westinghouse throughout the process.

2.3.4 Cross-cutting Topics

36 The SFP is a Cross-cutting Topic considered within this report.

2.3.5 Integration with Other Assessment Topics

37 Radiological protection interfaces with all the other assessment topics with the exceptions of electrical power supply systems, management of safety and quality assurance and security. The interfaces between the additional areas for Step 4 radiological protection assessment and other assessment topics were identified in Table 5 of the Step 4 Radiological Protection Assessment plan for the Westinghouse AP1000 (Ref. 1).

38 The interfaces with other assessment topics regarding normal operation are summarised below.

- Radiation sources: fuel design, reactor chemistry and radioactive waste and decommissioning.
- Designated areas (radiological classification of areas / radiological zoning): human factors and radioactive waste and decommissioning.
- Shielding: civil engineering, mechanical engineering and structural integrity.
- Contaminated Areas: reactor chemistry and radioactive waste and decommissioning.
- Ventilation: mechanical engineering and environmental issues.
- Radiological instrumentation: control and instrumentation, reactor chemistry and radioactive waste and decommissioning.
- Decontamination: reactor chemistry and radioactive waste and decommissioning.
- Optimisation for work activities (including fuel route): civil engineering, mechanical engineering, human factors and radioactive waste and decommissioning.
- Waste handling and decommissioning: radioactive waste and decommissioning.
- Public exposure from direct shine (direct radiation originating from within the site boundary): civil engineering and environmental issues (assessed by the Environment Agency).

39 The interfaces with other assessment topics regarding accident conditions are summarised below.

- Persons on-site: internal hazards, external hazards, PSA, fault studies, reactor chemistry and radioactive waste and decommissioning.
- Intervention personnel: internal hazards, external hazards, PSA, fault studies, reactor chemistry and radioactive waste and decommissioning.

- Off-site radiological consequence assessment (Level 3 PSA is reported in the Step 4 PSA Assessment of the Westinghouse AP1000 (Ref. 31): internal hazards, external hazards, PSA, fault studies, reactor chemistry and environmental issues.

40 In addition, an RO on source terms was raised jointly with reactor chemistry and radioactive waste and decommissioning assessors and another RO on decontamination, was raised jointly with radioactive waste and decommissioning assessors plus assessors from the Environment Agency.

41 In view of the interlinks between the disciplines of radiological protection and radioactive waste and decommissioning, ND undertook a series of site visits involving radiological protection and radioactive waste and decommissioning assessors to nuclear radioactive waste facilities operated by a range of companies across Europe (GB, France, Germany and Sweden) and the United States. Assessors from the Environment Agency joined ND on some of the visits. The purpose of these visits was to benchmark the design, layout and operation of nuclear radioactive waste facilities to assist us in the assessment of such facilities during GDA. My assessment report refers to the outcomes from those benchmarking studies as they relate to radiological protection.

42 Each benchmarking visit identified examples of relevant good practice with regard to radiological protection and / or radioactive waste and decommissioning. The full list of examples of relevant good practice are in Table 3 of the Step 4 radioactive waste and decommissioning report (Ref. 44). Examples relevant to radiological protection were as follows.

- Staged risk reduction based on pre-planned decommissioning stages is a good approach to decommissioning.
- Early consideration should be given to waste reduction, decontamination, segregation and recycling.
- International operational experience feedback should be actively sought when developing decommissioning methodologies.
- Robots have been developed and used for repetitive jobs in high dose environments, such as Steam Generator (SG) inspection and maintenance.
- Plant mock ups aid training and therefore reduce potential doses.
- A single fixed facility can operate effectively and deal with the waste from a number of reactors.
- To have confidence in the decommissioning approach, the plant needs to be characterised. For example, the operator needs to know the level of contamination in concrete and / or the background doses.
- The mapping of the radiological condition of the plant can take significant resource.
- The minimisation of Stellite^{TM} reduces doses to workers and appears to be practical from an engineering point of view.
- With a suitably shielded design, access into containment can be achieved with minimal dose.
- Space is needed in the waste management facilities to provide flexibility in dealing with the waste items a plant may produce over its operating life.

- The amount of space needed in the health physics laboratories needs to be sufficient to provide adequate separation between different activities, processes and samples.
- Where work is on a campaign basis, with long periods between campaigns, doses can be managed effectively by the use of a dedicated team who work frequently with the equipment on different sites.
- Contamination traps can be designed out of mobile decontamination machines.
- Items with high doses that require maintenance can be designed with quick release fixings.

43 These examples of relevant good practice assisted my assessment by demonstrating approaches to radiological protection that can be considered to be reasonably practicable for operators to implement. The examples cover many of the topics which were detailed in the Step 4 assessment plan (Ref. 1) such as radiation sources, designated areas, decontamination and optimisation of work activities, in addition to the design, layout and operation of nuclear radioactive waste facilities.

2.3.6 Out of Scope Items

44 The following items have been agreed with Westinghouse as being outside the scope of GDA.

- The design of robotic and remote handling technologies is out of scope, but AP1000 design features which facilitate their use are in scope.
- The design of temporary shielding is out of scope, but AP1000 design features which facilitate its use are in scope (for example, the provision of adequate space around radioactive components to permit the use of temporary shielding).
- The design of locally shielded items, such as glove-boxes and sample transport trolleys not associated with the generic reactor design is a site specific matter and is therefore out of scope.
- The selection of specific decontamination techniques is an operational issue and is out of scope, but the following considerations are in-scope:
  - Design aspects to facilitate decontamination.
  - Amount of space available for decontamination activities.
  - Potential doses from decontamination as related to design features.
- The use of portable (i.e. hand-held) radiological instruments is an operational issue and is out of scope. The specification of fixed radiological instrumentation (e.g. manufacturer) used as part of the RMS is out of scope.
- Doses associated with mid-loop working are out of scope. This topic will be assessed on a site specific basis should the licensee wish to undertake this practice.

3 WESTINGHOUSE'S SAFETY CASE

45 The December 2009 PCSR (Ref. 11) sets out the safety case for the radiological protection aspects of the AP1000 design. In isolation, this version of the PCSR did not contain sufficient information upon which to undertake an adequate assessment of the design. As a result, additional information was obtained from TQ and RO responses and this has enabled a sufficiently detailed assessment. These responses and additional documentation have been consolidated into the March 2011 version of the PCSR (Ref.12) and I am satisfied that this document is broadly consistent and adequately represents the information which has formed the basis of my assessment. As a result, I have referred to the March 2011 PCSR in this assessment report, and primarily to the information detailed in Chapter 24 which addresses radiological protection.

46 The PCSR covers the following key topic areas of radiological protection:

- Design targets.
- Radiological protection of workers – normal operation external radiation
  - Source term reduction;
  - Design features;
  - Plant layout;
  - Shielding.
- Predicted doses to workers.
- Radiological protection - members of the public.
- Control of surface and airborne contamination.
- Radiation monitoring.
- Operational health physics.
- Handling of radioactive waste.

47 I present information on Westinghouse's safety case for each of the topic areas in the relevant parts of my assessment in Section 4.

48 The appendices to Chapter 24 of the PCSR include extracts of dose estimates for several high dose tasks and tables which provide information on sources of radiation and the classification of areas.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR RADIOLOGICAL PROTECTION

49 The Step 4 Radiological Protection Assessment Plan (Ref. 12) identified a number of topics for assessment. These began with the source term, considered designation of areas (zoning classification), identified engineered features that influence radiation exposure (e.g. shielding), and followed up with optimisation of radiation exposure during work activities and accident conditions.

50 It was clear from the commencement of my assessment that additional technical expertise would be required from TSCs in order to support specific topics; namely shielding, criticality and high dose tasks. The fourth topic identified in my Step 4 assessment plan requiring TSC support, Plume dispersion modelling and dose consequences for Level 3 PSA, is now captured in the PSA assessment report for the AP1000 (Ref. 31). These topics have been the subject of detailed sampling and analysis, with the TSCs having reviewed a significant amount of supporting documentation provided by Westinghouse. My assessment of radiological protection has been supplemented by their analyses.

51 In order to obtain further information to support the claims outlined in the PCSR, I raised a number of TQs and ROs and the information provided by Westinghouse constitutes the arguments and evidence which have been used in my assessment. Several of the TQs and ROs were raised in collaboration with assessors in other topics areas. The TQs are in the Schedule of Technical Queries Raised during Step 4 (Ref. 8) and ROs are in the Schedule of Regulatory Observations Raised during Step 4 (Ref. 9).

52 A summary of Westinghouse's safety case, together with my assessment and its findings, are presented below with the exception of the assessment of criticality control in the SFP, which is reported separately in Appendix 1.

4.1 Normal Operation – Radiation Sources

4.1.1 Normal Operation – Radiation Sources - Assessment

53 The management of radiation sources associated with the operation of a nuclear reactor is a fundamental aspect of radiological protection at nuclear power stations. Since it is not normally practicable to eliminate the sources of ionising radiation, the emphasis must be on reducing the magnitude of the radiation sources in order to reduce radiation levels and correspondingly minimise the exposure of personnel and the public to ionising radiation. Although many measures, which can be taken to reduce radioactive sources associated with an AP1000 reactor, are related to the operating regime which is selected for the plant and so depend on the decisions taken by future licensees, there are aspects that are related to the physical design itself and these have been the subject of my assessment.

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

4.1.1.1 Assessment - Information on the Source Term

55 The PCSR (Ref. 12) provides a summary of the radiation sources associated with normal operation, maintenance and refuelling, including brief details of the source term, the

radiological significance relevant to radiological protection and measures which restrict the exposure of workers and the public. It segregates the potential effects on each of the exposed groups (i.e. workers and the public). A definitive list of source terms used for shielding assessments in support of the AP1000 design are presented in the Radiation Analysis Design Manual (Ref. 45).

56 I obtained additional information on the source terms associated with specific plant areas by raising several TQs (Ref. 8). This information was used in the shielding review carried out by a TSC, Nuclear Technologies, and constituted sampling by ‘deep slice review’. Topics covered by TQs included the calculation of neutron activation of materials, reactor vessel boundary source term generation, and source term code verification documentation. With the exception of the matters which were the subject of an RO (described below), information generally addressed my requirements and was provided by Westinghouse in a timely fashion.

57 The definition and appropriate use of the source term is an important stage in understanding and deriving the safety requirements of any nuclear activity. This source term often takes the form of a radioactive inventory plus any other parameters relevant to that particular nuclear activity. In the PCSR (Ref. 12) radioactive inventories are used in a number of different assessment areas and radioactive inventories may be manipulated to address specific purposes. For example, in some areas worst case inventories may be used, whereas in others it is more appropriate to use more realistic inventories.

58 ND and the Environment Agency recognised that while there was some consistency between the source terms used in different assessment topics, it was not always obvious how consistency was maintained and ND and the Environment Agency needed to understand the following points.

- How the radioactive source term had been derived.
- Justification for the overall suitability of the source term.
- Details of assumptions that could significantly affect the source term.
- Identification of assessments where the source term was used and how it was used.
- How the source term had been used consistently across the assessment areas.
- How the source term had been manipulated for use in each specific assessment area along with assumptions used.

59 The management of source term information is clearly a cross-cutting issue and so ND and the Environment Agency raised an RO (Ref. 9) jointly on source terms, which encompassed the disciplines of radiological protection, reactor chemistry, radioactive waste and decommissioning, best available technology, management of safety and quality assurance, PSA and fault studies.

60 The RO stated that ND and the Environment Agency did not consider that Westinghouse had shown how the source term had been derived, how the source term used was consistent across all assessment areas and how the source term was used in each specific assessment area. The RO action (ROA) required Westinghouse to demonstrate how these points were met and to identify the assessments where the source term was used (e.g. radioactive waste management, discharges, normal operations, accident conditions).

61 Westinghouse’s response to the RO (Ref. 9) stated that generic source terms, as detailed in the Radiation Analysis Design Manual (Ref. 45), are applied across a wide range of

operations and circumstances but where a source term needs to be modified, a specific calculation is completed and the methodology, assumptions and results are recorded in calculation notes. A single, small team carries out these calculations in order to promote consistency. In order to improve traceability of data, Westinghouse has created a source term reference matrix which serves as a "roadmap" to link each calculation note with its source term reference. This matrix is constantly being updated as more documents are generated.

62 The source terms are calculated using computer codes which are used for specific applications (for instance, FIPCO is used to determine fission product activity concentrations in the primary coolant, whilst CORA is used to evaluate the generation and transport of corrosion products in the Reactor Coolant System (RCS)). These codes are used by Westinghouse across many plants, but the inputs to these codes have been tailored specifically to the AP1000. Each calculation is independently verified by another team member to ensure its validity.

63 The competent development and application of source terms is a key consideration for radiological protection at the design stage of nuclear facilities, as they will form the design basis for shielding structures to protect personnel and the public. As part of a sample on this topic, Nuclear Technologies conducted a review of one calculation, noted 'AP1000 Radiation (neutron and gamma-rays) Sources for the Assessment of Streaming from the Reactor Cavity' (Ref. 46) and judged that the calculations used to generate the boundary source term have been undertaken using acceptable codes, methods and cross-section data, when compared against relevant good practice in the UK. In the context of shielding design, the TSC concluded the following:

- That no concerns were raised during the review of the source terms used in defining the shielding provisions for the AP1000.
- That source terms are well defined and as a result of key assumptions it is apparent that they will be conservative when compared with more realistic source terms based upon observations on existing plants.
- That it has confidence that the shielding provisions and the predicted dose rates for any given area of the plant will also be conservative with respect to protection of the public and personnel from external radiation.

64 Further details of the assessment of the assessment of the shielding source term are provided in Section 4.3.1.2.

4.1.1.2 Summary - Information on the Source Term

65 Westinghouse has provided a safety case that satisfies regulatory expectations regarding derivation of source terms, identification and justification of assessments where the differing source terms have been applied and consistency of application of source terms across assessment areas.

4.1.1.3 Assessment - Reductions in the Source Term

66 Reducing the source term associated with nuclear plant can significantly reduce worker exposure without the need for applying systems, which are further down the hierarchy of control measures, such as engineering or administrative controls. As a result, minimising the source term should be a fundamental radiological protection consideration of designing new nuclear power stations.

67 Expectations for reductions of source terms are covered in Para. 479 of the SAPs (Ref. 4) in that there should be a strategy to restrict radiation exposure. This strategy should include, amongst other things, the minimisation of sources of radiation. RP.1 also states that adequate protection against radiation exposure should be provided in those parts of the facility where access needs to be gained. The need to minimise sources of radiation is also emphasised in Para. 4.1 in the TAG on Radiological Protection (Ref. 16) where it advises that consideration should be given to minimising the formation of activated corrosion products from circuit components.

68 Chapter 24 of the PCSR (Ref. 12) claims that source terms have been minimised in the AP1000 design using the following means:

- Material selection.
- Material quality control.
- Piping design.
- Reactor coolant chemistry control.
- Primary coolant and SFP cleanup.
- Surface treatment of the SG channel head.

69 The ND assessment of Westinghouse's efforts to reduce source terms has principally been led by the reactor chemistry assessors and their conclusions are outlined in the step 4 assessment report (Ref. 47). I have liaised closely with these assessors on this matter and a summary of our key conclusions are outlined below. I have principally focused on radionuclides, which constitute an external radiation hazard but doses resulting from sources of internal radiation exposure are described in Section 4.4.1.1.1.

4.1.1.3.1 Material Selection and Design Features

70 In many Nuclear Power Plants (NPP), two radioisotopes of cobalt, cobalt-60 and cobalt-58, typically account for over 80% of equivalent dose rates associated with the primary coolant. Cobalt-60 occurs through the neutron activation of cobalt-59, while cobalt-58 occurs through the activation of nickel-58. There are three principal sources of activated cobalt:

- Corrosion products from components made from cobalt or high cobalt alloy.
- Corrosion of steels and alloys which contain traces of cobalt.
- Corrosion of nickel alloys with subsequent activation.

71 High cobalt alloys have had particular use as hard wearing alloys (e.g. Stellites™) and are commonly used in PWR components such as Control Rod Drive Mechanisms (CRDM), valve seats and wear pads, where this property is desirable. Once Stellites™ were identified as significant contributors to worker dose, much work was undertaken around the world to progressively eliminate Stellite™ from various components, This strategy has been successful in reducing worker doses at German 'Konvoi' sites and demonstrates good practice in reducing the cobalt source term at Pressurised Water Reactors (PWR). It is acknowledged, however, that 'Konvoi' and AP1000 designs have differing intended plant lifetimes and so any reductions in Stellite™ would need to be compatible with the ability of the plant to withstand the additional period of operation.

72 Westinghouse has been conducting an extensive programme, reviewing the use of Stellites™ in CRDMs, the remaining valves and other components. Westinghouse has tested Norem™ as a low-cobalt replacement material for Stellite™ in certain components and has found it to be effective in certain components. However, Westinghouse has taken a guarded approach to replacing these components in the reference design, although there is obvious scope to change this position during the site specific phase (Ref. 47).

73 Large numbers of isolation valves in earlier PWRs were constructed using Stellite™ and Westinghouse has stressed that the design of the AP1000 utilises fewer valves than a conventional PWR and so starts from a position of containing less cobalt. Reactor coolant pump bearings contain Stellite™ and so are another potential source of cobalt. The reactor chemistry assessors have determined that it would be difficult to further reduce the level of wear in these bearings, but the fact that this source of cobalt is outside the core means that other chemistry measures, such as Chemical and Volume Control System (CVS) purification and addition of zinc, can be used to optimise cobalt removal (Ref. 47).

74 The PCSR (Ref. 12) states that use of hard-facing material with cobalt content such as Stellite™ is limited to applications where its use is necessary for reliability considerations. It is noted that the design must reach a balance between reducing high cobalt materials in order to restrict worker doses, and ensuring that components are robust enough to withstand operational conditions, since their failure could lead to increased exposure for personnel during repair. I am satisfied that Westinghouse has taken adequate steps to minimise the use of hard-facing cobalt alloys in the AP1000 and have undertaken development work on minimising their use further.

75 In order to further minimise the cobalt-60, cobalt-58, iron-55 and nickel-63 inventory of the AP1000, Westinghouse has specified the following measures with regard to materials selection (Ref. 12):

- Corrosion resistant cladding material will be specified in order to prevent the ferritic low-alloy and carbon-steel construction materials used in the Reactor Pressure Vessel (RPV) and other pressure-retaining applications from contact with reactor coolant.
- Low-cobalt tubing material (<0.015%) is specified for the AP1000 SG design. The large surface area associated with the SGs mean that this measure will reduce the quantity of cobalt in the primary circuit, resulting from corrosion and is likely to decrease dose rates for work on the SGs and the primary circuit as a whole.
- Nickel-based alloys in the RCS are used only where component reliability may be compromised by the use of other materials. Most of the nickel will be associated with Inconel 690, used in the SG tubing, tube divider plates and radial supports and nozzles in the reactor head. Material specified for the AP1000 will have a higher chromium content to minimise corrosion of nickel.
- The prevention of contact between lead, antimony, cadmium, indium, mercury and tin and the component parts of wetted surfaces made of stainless steel or high-alloy materials during fabrication or operation. In addition, bearing alloys containing these metals are not used in contact with primary coolant.

76 While cobalt impurities have been minimised in certain primary circuit components, Westinghouse has not given the same commitment for all components which come into contact with primary coolant. As a result, some of the dose rates associated with areas

within containment appear to be excessive (as discussed in Section 4.2.1.1.3.). It is not clear whether Westinghouse actually plans to use components with these specifications, or whether they were simply used as assumptions for shielding assessments (which are typically conservative) in order to provide a 'worst-case' scenario. Additionally, although dose rates in certain areas might appear high, there is no evidence that this matter will lead to excessive doses to workers, with the possible exception of SG work (described in Section 4.4.1.3.2). However, in order to demonstrate ALARP, the operator will need to demonstrate that cobalt and other elements, which may become activated and contribute significantly to operator radiation exposure, have been reduced so far is reasonably practicable in the site specific design and so this matter is the subject of Assessment Finding **AF-AP1000-RP-01**:

> ***AF-AP1000-RP-01:*** *The licensee shall provide a report which demonstrates that the content of cobalt and other elements within primary circuit materials which may become activated and contribute significantly to operator radiation exposure has been reduced so far as is reasonably practicable. The report shall take into account improvements that Westinghouse has identified, in addition to new materials which may have become available following the GDA process. This finding shall be addressed before mechanical, electrical and C&I safety systems, structures and components are delivered to Site.*

77 Successful reductions in cobalt-60 will leave proportionally more cobalt-58 because it is more difficult to replace Inconel than Stellite™ in reactors. The Inconel alloys used in most PWRs, especially for SG tubing, contain a significant proportion of nickel. Any slight corrosion of such a large area will result in some nickel transferring to the core of the reactor. Alloy 800 is not an Inconel and could be a replacement, but this would be a significant change to the whole design (Ref. 47).

78 To minimise corrosion, Inconel alloys contain chromium, which is meant to form a corrosion resistant layer. Unfortunately, the chromium level in the Inconel 600 used in earlier PWRs was insufficient to prevent chromium-free regions developing at grain boundaries during thermal treatment (Ref. 47). As stated above, the Inconel 690 specified for AP1000 has much higher chromium content, which helps to stabilise the grain boundaries. Steam-generator tubing, tube divider plates and some radial supports and nozzles in the reactor head will be made of Inconel 690 (Ref. 47).

79 Many PWRs using Inconel 690 have benefitted from low radiation fields (Ref. 47). However, actual radiation fields can depend on finishing treatments applied to metal components by their manufacturers and to the commissioning carried out. In order to minimise general corrosion, Westinghouse is improving methods of finishing and conditioning surfaces, particularly of SG tubing and by electropolishing the channel heads and water-chambers in the SGs. Further information on the assessment of Westinghouse's arrangements in this topic area is detailed in the Step 4 reactor chemistry assessment of the AP1000 (Ref. 47).

80 The AP1000 has the following design features associated with piping in order to reduce the source term associated with crud deposits:

- Pipe bends are used rather than elbows, where practicable, in order to reduce potential crud traps.
- Welds are made smooth to avoid crud traps.

- Piping connections to the reactor coolant loops are located on or above the horizontal centreline of the pipe, wherever practicable, in order to minimise crud build-up in branch lines resulting from gravitational separation.

81 I also raised a TQ to request information from Westinghouse on its measures to reduce the direct activation of materials, which are exposed to the high neutron flux (Ref. 8). Westinghouse has already specified a low cobalt alloy for the RPV (Ref. 47) and the TQ response highlighted several design features intended to reduce the activation of structures and components external to the core. These include a carbon steel liner and structural stiffeners to reduce activation of the bioshield, titanium and lead shields for source range detectors and borosilicone neutron blocks, which reduce upward streaming of neutrons from the reactor cavity. From the information provided, I have no reason to believe that Westinghouse has not taken all reasonably practicable measures to reduce the direct activation of materials which are exposed to the high neutron flux.

4.1.1.3.2 Reactor Coolant Chemistry Control

82 The effectiveness of the materials selection and design features outlined previously will depend upon the chemical control of the RCS. Westinghouse’s approach is to reduce corrosion and crud formation, thereby reducing the radioactive inventory associated with the coolant.

83 General corrosion is greatest at the beginning of a reactor’s life and should fall by a large factor over 4 – 8 cycles of operation (Ref. 12). This process is known as passivation and the target for a good reactor is to achieve a high factor of improvement in the least number of cycles. Passivation is considered in detail in the reactor chemistry assessment report (Ref. 47).

84 The PCSR (Ref. 12) states that the RCS water chemistry will be selected to minimise corrosion. Chemicals will be added via the CVS which are intended to control pH, scavenge oxygen and control radiolysis reactions in the coolant, whilst controlling suspended solid and impurity concentrations.

85 Zinc injection into the RCS has been identified as a key tool in reducing dose rates associated with the primary circuit by thinning and stabilising corrosion films on internal surfaces which are in contact with primary coolant, thereby reducing corrosion of materials.

4.1.1.3.3 Primary Coolant and Spent Fuel Cleanup

86 The CVS utilises filters and ion exchange resins in order to reduce the levels of corrosion and ionic fission products present in the primary coolant. The purification of reactor coolant reduces the exposure of workers during outages. Similarly, the Spent Fuel Pool Cooling System (SFS) cleans up water from the SFP, the refuelling cavity and the in-containment water storage tank (IRWST) by filtration and ion exchange. A key principal of minimising the source term associated with the AP1000 will be fuel reliability, with an emphasis being placed on the avoidance of fuel defects and failures. When calculating source terms for shielding design, Westinghouse has adopted a conservative assumption on the number of fuel failures when, in reality, the number of failures should be significantly lower than in previous generations of PWRs (Ref. 47).

87 Assuming that the levels of failed fuel will be low, the principal source of external radiation in the SFP will be cobalt-58 and cobalt-60 associated with crud. These deposits of

corrosion products will be transferred to the SFP, when fuel assemblies are offloaded during outages and some crud will become re-suspended in the water during the course of fuel handling. Westinghouse claims that the SFS clean up rate is much higher than in previous generations of PWRs, with two water volume changes during each 24 hour period and this should contribute to the minimisation of radioactivity in the SFP (Ref. 47).

4.1.1.3.4 Surface Treatment of the Steam Generator Channel Head

88 Electropolishing can reduce the accumulation of radioactivity on out of core surfaces. In the standard form of electropolishing, the metal surface is given a positive charge in contact with a conductive solution. A high electric current then polishes the surface, reducing activity uptake by a factor of two. Westinghouse considered that electropolishing the channel heads and water-chambers in the SGs was justified in AP1000 (Ref. 47). This demonstrates an ALARP approach to dose-reduction in the design of AP1000. Although electropolishing of SGs has been adopted into the generic design, the dose estimates described in Section 4.4.1.3.2 do not currently take account of electropolishing of the SG Bowls and so doses discussed in that section are conservative and should be lower during operation.

4.1.2 Normal Operation – Radiation Sources - Conclusions

89 I have determined that Westinghouse has carried out a detailed assessment of the source terms associated with the AP1000. Westinghouse has provided evidence of its strategy to minimise these source terms as part of the generic design and I am broadly satisfied with the measures which it has outlined. However, it appears that there is scope for further reductions at the procurement stage, particularly with regard to cobalt impurities in primary circuit components.

4.1.3 Normal Operation – Radiation Sources - Findings

90 Westinghouse has demonstrated efforts to select materials so as to reduce the radiation source term associated with the AP1000. There is scope for improvements with regard to materials selection at the procurement stage, which will further reduce the source term and so the following Assessment Finding has been raised:

> ***AF-AP1000-RP-01:*** *The licensee shall provide a report which demonstrates that the content of cobalt and other elements within primary circuit materials which may become activated and contribute significantly to operator radiation exposure has been reduced so far as is reasonably practicable. The report shall take into account improvements that Westinghouse has identified, in addition to new materials which may have become available following the GDA process. This finding shall be addressed before mechanical, electrical and C&I safety systems, structures and components are delivered to Site.*

4.2 Normal Operation – Designated Areas

91 IRR99 require areas to be designated as controlled or supervised areas (Ref. 19) and the designation within a nuclear facility, should take account of the level of hazard and risk from exposure to external radiation and / or internal radiation from surface and airborne

contamination. Such a designation scheme is usually referred to as a radiological classification of areas scheme in the UK and as a radiological zoning scheme in the US.

92 The objectives of a radiological classification of areas scheme are as follows.

- To ensure compliance with legal requirements.
- To assist in the control of radiation dose uptake (through both external and internal exposure).
- To enable a consistent and efficient plant layout to be developed as a useful basis of design.

93 In general, the shielding provisions for a proposed nuclear facility are initially based on a preliminary classification of areas scheme, which outlines the upper bound dose rates within the scheme for each room of the facility, based on the expected occupancy requirements for activities to be undertaken in that room.

94 The radiological classification of areas document is generally considered to be a live document, which is revised as required throughout the design and operational phases of the nuclear facility. In general, the shielding provision for nuclear facilities designed in the UK are initially based on design targets outlined in a preliminary radiological classification of areas scheme, developed with guidance sought from plant operators, design engineers, and radiation protection advisers.

95 The topic of designated areas is of concern to the radiological protection of workers, rather than protection of the public. Section 4.3.1.6.1 of the shielding assessment has confirmed that external radiation doses to the public are negligible under normal operations.

4.2.1 Normal Operation – Designated Areas - Assessment

96 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of designated areas (radiological classification of areas / radiological zoning) would include the following matters.

- Zoning for levels for direct radiation, surface contamination and airborne contamination.
- Control of access by engineered controls and managerial controls.
- Optimisation of access and egress routes.

97 RP3 and Para. 485 of the SAPs (Ref. 4) on designated areas advise that further division of designated areas should be based upon the levels of radiation, contamination and airborne activity, measured and / or expected as a result of particular planned work activities. The designated areas should also have associated controls to restrict exposure and prevent the spread of radioactive substances.

98 Paras 4.6 and 4.7 of the TAG on Radiological Protection (Ref. 16) advise that the zone category should indicate the required degree of engineered and managerial controls and should increase for increasing levels of radiation and contamination, e.g. R1, R2, R3, etc. and C1, C2, C3, etc. for increasing levels of radiation and contamination, respectively. In addition, access to the facility control room and other low radiation areas with high occupancy should not require access through zones that would require substantial precautions. Also, higher category zones should be nested within less highly categorised zones.

4.2.1.1 Zoning for External Radiation

99 Chapter 24 of the PCSR (Ref. 12) defines Westinghouse's zoning strategy for external radiation and an extract of this information is included below. Westinghouse has also interpreted the zoning into a context which would be familiar to the UK. It should be noted that the designation scheme described by Westinghouse, has been developed for design purposes only and a different system of designating areas is likely to be derived by an operator at the site specific phase.

Table 1: Westinghouse's AP1000 Zoning Criteria for Radiation

Designation	Maximum Design Dose Rate	Access	Typical UK Area Designation Arrangements
0	≤0.5 μSvh^{-1}	Unlimited general occupancy	Undesignated area
I	≤2.5 μSvh^{-1}	No restriction on access	Supervised area
Restricted Radiation Zones			
II	≤25 μSvh^{-1}	Occupational access	Further subdivision for normal operations Areas >2.5 µSv/h subject to limits on continuous occupancy
III	≤150 μSvh^{-1}	Periodic access	Access restricted to 1 to 10 hours per week
IV	≤1 $mSvh^{-1}$	Limited access	Access restricted to <1 hour per week
V	≤10 $mSvh^{-1}$	Controlled access	Access restricted to a few hours per year at most
VI	≤100 $mSvh^{-1}$	Normally restricted Post-accident: limited	No access during normal operations Limited post-accident access
VII	≤1 Svh^{-1}	Normally severely restricted Post-accident: restricted	No access during normal operations Very limited post-accident access
VIII	≤5 Svh^{-1}	Normally inaccessible Post-accident: severely restricted	No access during normal operations Extremely limited post-accident access
IX	>5 Svh^{-1}		No access

100 Westinghouse has described the designation of each room in the Containment, Auxiliary Building and Annex Building for operational, shutdown, and accident conditions (Ref. 12) based on external radiation. This information provides a useful radiation profile of a generic AP1000 facility and provides confidence that Westinghouse has made considerable effort to understand the external radiological hazards associated with each plant area. By comparing this area zoning with intended access requirements, Westinghouse has been able to focus its efforts, with regard to the design of shielding, on areas with increased radiological risks in order to strive to reduce doses so far as is reasonably practicable.

101 It should be noted that in some cases Westinghouse has allocated a designation of Zone II (≤25 µSvh^{-1}) to high occupancy areas where I would expect the dose rate to be negligible, such as in offices and other areas at the entry to the RCA in the Annex Building. In its response to an RO on health physics facilities, Westinghouse has confirmed that the assumptions used for these classifications have often been conservative and, in reality, the dose rates in these areas should be at or near environmental background dose rates (Ref. 9), which meets my expectations.

102 A review of Westinghouse's zoning scheme was carried out by a TSC, TÜV SÜD. The TSC has wide experience of radiological protection at German PWRs including 'Konvoi' plants, which have some of the lowest radiation doses in the world.

103 The TSC focused its assessment on the radiation zoning applied to the Containment Building during normal operation and 24 hours after shutdown. The principal documents reviewed during this assessment were Refs 48 and 49. The TSC compared Westinghouse's area designation scheme with UK regulatory requirements and international relevant good practice, including a comparison of like-for-like dose rates at German PWRs. Its observations are discussed below.

4.2.1.1.1 Zoning - Containment Building at Power

104 The TSC concluded that the designation of all areas within the Containment Building as controlled areas is appropriate. When the reactor is operational, the majority of areas being examined in Ref. 48 have been allocated an initial designation falling into the range of zones V to VII. Some calculated dose rates have exceeded the upper limit of their allocated zone by a large margin. The area with the highest dose rate, which has been identified as exceeding its allocated zone criteria, is the Lower Pressuriser Compartment (with a stated dose rate of 330 mSvh^{-1} exceeding the upper limit of 100 mSvh^{-1}). Other affected areas include part of the Maintenance Floor (dose rate 2.1 mSvh^{-1} exceeding upper limit of 0.15 mSvh^{-1}) and a section of the Refuelling Cavity (calculated dose rate of 16 mSvh^{-1} exceeding upper limit of 1mSv/h). Ref. 48 is not clear whether exceeding the upper limit will impact doses, since no information is provided on whether any work is required in these areas at power (see Section 4.2.1.1.2 below for further discussion on this topic).

105 Calculated dose rates for other areas are significantly lower than the allocated zone limits, such as those in the SG Compartments on the operating deck (calculated dose rates of approximately 65 mSvh^{-1} as opposed to an upper limit of 1 Svh^{-1}). This means that the designated zone could potentially be decreased.

106 The main contribution to the dose rate in the SGs, primary coolant legs and Reactor Coolant Pumps (RCP) and Pressuriser and Surge Line is nitrogen-16 and crud associated with the primary circuit, while the main contribution to dose rates in the

Reactor Cavity is from radiation streaming from the RPV. Contributions from the filters, demineralisers and heat exchangers in the CVS and backscatter from the containment vessel is also considered.

107 The primary coolant in the Surge Line and Hot and Cold Legs is identified as the principal cause of dose rates, which have led to a dose rate in excess of zone limits. Two strategies for reducing dose rates involve reducing the source term of the primary coolant or improving the shielding in certain areas.

108 Westinghouse has identified an SG Compartment wall as being one area where the shielding could be optimised and suggests that the conservative assumptions, used to generate the source term for the Surge Line, could be revisited. The TSC has recommended that improving the shielding around the Surge Line could improve dose rates and also recommends reducing cobalt-60 impurities in stainless steel components in order to reduce crud films in the primary circuit. Efforts have been made by Westinghouse to reduce the cobalt content of steel in the SGs to 0.015%, but if cobalt content is as high as 0.25% in components which come into contact with primary coolant, then there remains a high potential for corrosion and activation of cobalt to lead to enhanced dose rates. Assessment Finding **AF-AP1000-RP-01**, detailed in Section 4.1.2, addresses the reduction of cobalt content in primary circuit components.

109 The adequacy of shielding associated with high dose rate components, such as the Surge Line, would be expected to be substantiated as part of a future licensee's response to Assessment Findings **AF-AP1000-RP-02** and **AF-AP1000-RP-03**, as described in Sections 4.2.1.1.2 and 4.2.1.1.4, respectively.

4.2.1.1.2 Access – Containment at Power

110 With the reactor operational, many areas within the Containment have calculated dose rates in the range of tens-of-millisieverts per hour (with several exceeding 100 $mSvh^{-1}$) which will challenge the radiological safety of workers. The TSC asserts that it should be reasonably practicable at modern PWRs to restrict individual doses to below 5 mSv and in many areas this value would be exceeded in a matter of minutes.

111 Although Westinghouse has specified that the areas with the highest dose rates, including the upper RCP Areas of the SG Compartments (calculated dose rates >200 $mSvh^{-1}$) (Ref. 12), can only be accessed during reactor shutdown, the access restrictions for other areas are not specified. This was an area of concern and so was the subject of further assessment.

112 The Annual Occupational Dose Evaluation (Ref. 50) states that 100 worker-hours per year of containment access is associated with the AP1000 plant during at-power conditions. A list of potential operations (as outlined in Ref. 50) which may be required whilst at-power for an AP1000 site is also provided in Table 2 below (reproduced from Ref. 50):

Table 2: Potential Operations requiring Access to Containment Whilst At-power

Surveillance
• Routine Patrols
• Health Physics Surveys
• Patrols to Identify System Leaks
Mechanical Repairs
• Isolate System/Component Leaks
• Valve Adjustments/Repairs
• Leak Test Personnel Hatch
• Fan Cooler Repairs
• RCP Oil Additions
• Repair In-Core Detector Drive System
• Repair/Replace Radiation Monitoring Detectors
Electrical Repairs
• Repair/Replace Transmitters
• In-Core Thermocouple System Repairs at Junction Box
• Valve Operator Repairs

113 A TQ on this subject had been raised in Step 3 (Ref. 8) but, during discussions with Westinghouse at its office in Pittsburgh, it transpired that new information may have emerged as a result of feedback obtained from US PWR operators. I raised a TQ (Ref. 8) in order to obtain the following information:

- Details on the types of activities which might be carried out in the Containment Building whilst at power, including working locations, types of workers exposed and expected dose rates.
- Details on the updated dose estimates associated with entry to the Containment Building whilst at power.

114 The TQ response confirmed that an informal benchmark of several operating reactors in the US had been conducted. The reasons and frequencies for accessing containment at power included:

- Entries are made for collecting safety injection tank samples and to perform venting operations each month, two to three times for oil additions to RCPs and just before an outage to perform any pre-job setups outside the bio-shield.
- The most frequent routine maintenance tasks performed are lubrication of a specific fan motor bearing and Personnel Airlock Door pressure test. Average is 5-6 power entries per unit per year.

- Routine entries are made quarterly. Other emergent work is scheduled as needed.
- Less than 20 entries per year are made to containment while the plant is operating at power (for emergent work).

115 Westinghouse claim that the following design futures of the AP1000 preclude the need to enter containment at power (Ref. 8):

- The use of seal-less reactor coolant pump motors has eliminated the need for an oil lubrication system.
- Samples from tanks within containment are not required during operation.
- There are no fan motors within containment requiring routine maintenance.

116 Westinghouse notes that a future licensee may still choose to enter containment at power for emergent work or to prepare for an outage. Whilst it is clear that the AP1000 has been designed to minimise these requirements, the high dose rates associated with some plant areas at power mean that this topic is the subject of an Assessment Finding (**AF-AP1000-RP-02**) which requires a licensee to identify and justify all likely work which is to be undertaken whilst at power.

> ***AF-AP1000-RP-02:*** *The licensee shall identify, and provide a justification for, all reasonably foreseeable work activities that are likely to require entry to the Containment whilst at power. For each of these activities, the licensee shall justify the reasons for this exposure, assess likely worker doses, and substantiate whether doses have been reduced so far as is reasonably practicable. This finding shall be addressed before fuel on-site.*

4.2.1.1.3 Zoning – Containment 24 Hours after Shutdown

117 The 'In-Containment Radiation Zoning at 24 Hours After Shutdown' (Ref. 49) document outlines dose rates for each plant area within containment and describes the principal source terms which contribute significantly to radiation exposure.

118 The highest designation allocated is Zone VI and applies to the Lower Reactor Cavity. The highest dose rate listed is 91 $mSvh^{-1,}$ which is slightly below the 100 $mSvh^{-1}$ limit for that zone. The dose rate is essentially due to the neutron activation of the concrete walls and floor and of the stainless steel panels of the Reactor Vessel Insulation System (RVIS) lower shell. The dose rate is dominated by the contributions of sodium-24 in the concrete and by cobalt-60 in the stainless steel. The short half-life of sodium-24 means that the dose rate is dominated by Cobalt-60 (half-life of 5.3 years) in the mid and long-range period.

119 Certain areas of the Nozzle Gallery exceed their allocated zone, which is a Zone V, with dose rates as high as 41 $mSvh^{-1}$ which exceeds the upper zone limit of 10 $mSvh^{-1}$. High neutron fluxes in this area lead to the activation of cobalt impurities in stainless steel materials to cobalt-60 and the main components, which contribute to the high dose rate, are Lower Neutron Block (LNB) liner, the Reactor Vessel Insulation System (RVIS) and the Hot and Cold Legs. The corrosion products deposits and primary coolant activity mainly contribute to the total dose rate in the vicinity of the primary loop piping.

120 The majority of the SG Compartment is designated as Zone V, with the principal contribution to the dose rate arising from corrosion products and primary coolant. The calculated dose rates meet this designation, with the highest dose rate being 4.3 $mSvh^{-1}$.

121 The highest zone allocated to the Pressuriser Compartment is Zone V and the upper limit is exceeded in one area; the Lower Pressuriser Compartment, which has a dose rate of 18 $mSvh^{-1}$. The enhanced dose rates result from the contribution from the Surge Line, due to activated corrosion products.

122 The Operating Deck and the Maintenance Floor and its mezzanine are allocated Zone II dose rates, with upper limits of 25 μSvh^{-1}. The calculated dose rates meet this designation criteria with one exception, the area outside the SG1 access room over the access tunnel grating, which has a calculated dose rate of 570 μSvh^{-1}.

123 The TSC that carried out a review of the zoning documents has extensive experience of working on German PWRs and was asked to compare the estimated dose rates for an AP1000 reactor to German reactors. The TSC concluded that the calculated dose rates associated with the AP1000 were higher than German reactors (Ref. 41). Dose rates in the Pressuriser Compartment and SG Compartments were described as being higher than the older generation of German PWRs and significantly higher than the latest generation 'Konvoi' plants, which have external dose rates which illustrate what could be taken to represent relevant good practice. It also notes that the dose rates in some areas would seriously restrict occupancy during outages.

124 Since IRR99 requires that dose sharing should not be the primary means of restricting exposure, it is reasonable to expect that these dose rates are minimised at the design stage. Westinghouse has made it clear that the dose rates presented have been calculated using conservative assumptions and methods, such as assuming a relatively high number of fuel defects and high cobalt impurities of 0.25% in many stainless steel components. WEC argue that the actual dose rates in an operating AP1000 would be lower than those presented in submissions to date, and I consider this to be a reasonable claim.

125 Westinghouse has carried out a sensitivity analysis on the cobalt content of steel components in order to minimise dose rates associated with activation products (Ref. 49). It is clear that further reductions in cobalt impurities would yield benefits in reducing dose rates further and the TSC suggests that levels could be reduced to a maximum of 0.05% in components and systems which come into contact with the primary coolant. In order to ensure that improvements are enacted, Assessment Finding **AF-AP1000-RP-01** has been raised on cobalt content of materials (see Section 4.1.3).

4.2.1.1.4 Access - Containment Building 24 Hours after Shutdown

126 The reactor will be shut down in order to allow access for scheduled outages (e.g. refuelling and maintenance) and potentially for emergent work such as repairs. The exposure of personnel undertaking this work is described in Section 4.4.1.3.

127 When determining the zoning for an area and hence the control measures which will need to be employed in order to ensure that the radiological conditions are consistent with that zoning, the intended occupancy and the nature of the work to be undertaken in that area should have been considered. With some aspects of the AP1000 submission, it is difficult to relate the radiation zoning of each area to the type of work which may be undertaken in that area. As a result, it is difficult to assess whether Westinghouse has taken all reasonably practicable steps to incorporate engineering controls, such as effective localised shielding around high radiation components, in order to minimise external radiation dose rates and resultant worker doses.

128 I am satisfied that Westinghouse has employed a zoning strategy which is based on likely access requirements for workers and that it has recommended engineering controls such as shielding in order to achieve external radiation levels consistent with that zoning. However, I have not had complete visibility of the exact work requirements for all controlled areas. This matter is of greater concern for high dose rate areas, such as those within containment. This is to be expected at the generic design stage because specific work programmes will not be finalised until the site specific phase. When access requirements are finalised, an ALARP justification will need to be undertaken for external radiation levels in accessible areas to demonstrate that all reasonably practicable measures have been taken to restrict the exposure of personnel to ionising radiation. The necessary scope of such an ALARP justification is likely to increase as the radiation zoning (and hence the level of radiological risk) increases.

129 In order to ensure that this ALARP justification is completed prior to the operation of an AP1000 reactor site, this matter has been captured as an Assessment Finding (**AF-AP1000-RP-03**).

> ***AF-AP1000-RP-03:*** *The licensee shall, taking into account any changes to the radiation source term and shielding design which have been made since GDA, provide a report that identifies external dose rates for all controlled areas during normal operation. In addition, the licensee shall submit an ALARP justification for areas to which access is required and where the dose rate exceeds 150 micro-sieverts per hour (during normal operation) to demonstrate that dose rates have been reduced so far as is reasonably practicable. The justification shall consider the nature of the work which is likely to be conducted within those areas, including the magnitude and duration of the exposure, and the number of workers exposed. This finding shall be addressed before fuel on-site.*

4.2.1.2 Zoning for Contamination

130 Westinghouse has utilised a colour-coding system for designating areas based on airborne and surface contamination. This classification system has been derived in order to assist in the design of ventilation systems, rather than for operational contamination control purposes. An extract of the PCSR is provided in Table 3 below (reproduced from Ref. 12):

Table 3: Westinghouse AP1000 Zoning Criteria for Contamination

Area Classification	Description		
White	Clean area free from radioactive contamination, whether surface or airborne.		
Green	An area which is substantially clean. Only in exceptional circumstances is airborne contamination such that provisions must be made for its control.		
Amber	An area in which some surface contamination is expected. In some cases there will be a potential for airborne contamination such that provision must be made for its control.		
Red	An area in which contamination levels are so high that there is normally no access without appropriate respiratory protection.		
Area Classification	**Surface Contamination Bq.cm^{-2} βγ**	**Airborne Activity Derived Air Concentration (DAC)**	**Typical Area**
White	<4	<0.01	Non-active areas
Green	Usually <4	<0.03	Most of the RCA during normal operations.
Amber	Possibly >4	<0.1	Some parts of containment during shutdown e.g., around SGs and Reactor Vessel Closure Head (RVCH) stand.
Red	>4 expected	>0.1	Flask pit before and during decontamination Vessel and pump rooms during breach of containment.

131 Chapter 24 of the PCSR (Ref. 12) states that only beta/gamma radiation is likely to be detected in areas where contamination is present. If present, levels for alpha surface contamination would be 10% of beta/gamma limits.

132 Westinghouse claims that the Radiologically Controlled Area (RCA) specified as Zone II in the PCSR forms the boundary of the areas with potential for contamination. Within the Radiologically Controlled Area (RCA), some areas will have higher potential for surface and airborne contamination particularly during refuelling and maintenance. Access control and traffic patterns are considered in the plant layout to reduce the spread of contamination.

133 The design considerations which affect the spread of contamination, such as the hierarchy of contamination control measures, are discussed in Section 4.5.

4.2.1.3 Optimisation of Access and Egress Routes

134 Areas of the auxiliary/annex buildings with negligible radiological risk are completely segregated from the RCA. A single personnel entrance is provided for the RCA of the reactor containment, Shield Building, Auxiliary Building, Annex Building and Radwaste Building. Access into the RCA, the non-active areas of the auxiliary building and the

turbine building is limited to authorised personnel via turnstiles. Entries are recorded, enabling rapid roll calls in emergencies. This approach is intended to ensure that access is restricted to authorised personnel only and that workers are subject to contamination control measures.

135 Westinghouse claims that the plant layout is such that access to a given radiation zone does not generally require passing through a higher radiation zone (Ref. 12). However, Westinghouse notes that in the case of an abnormal occurrence or accident, the zone restrictions may change because of increased dose rates. I have not assessed access and egress routes in depth, but the information provided by Westinghouse has given me no cause for concern.

4.2.2 Normal Operation – Designated Areas - Conclusions

136 By identifying and applying a radiological zoning scheme to the AP1000 design and then assessing whether the external radiation criteria are met, Westinghouse has created a sufficiently detailed radiological profile of the facility. It has expended significant effort in quantifying the external radiation hazard associated with the AP1000 and, as a result, has provided confidence that Westinghouse understands the hazards associated with the AP1000 design.

137 Although calculated using conservative assumptions, there are areas within the Containment where dose rates are extremely high. These have generally been identified by Westinghouse and recommendations have been put forward to decrease the dose rates further, by reducing the source term or incorporating localised shielding. Although these dose rates are of concern, the improvements that have been highlighted are associated with the detailed design and the procurement of components rather than being symptomatic of flaws with the bulk shielding design. As a result, these matters are captured as assessment findings rather than GDA Issues.

138 The topic of cobalt content in materials associated with the primary circuit is the subject of an Assessment Finding in Section 4.1.3. Additional Assessment Findings have been raised in order to address concerns over access to the Containment at power and the ALARP justification of external dose rates in controlled areas with regard to planned work to be undertaken in those areas. It should be noted that the concerns over enhanced dose rates within containment are related to the minimisation of worker doses alone; the shielding associated with the civil structure of the Containment ensures that the external radiation risk to the public from these sources under normal operations would be negligible (see Section 4.3.1.6.1).

139 Westinghouse has outlined a scheme for zoning for airborne and surface contamination which has been primarily used for ventilation design, rather than operational radiological protection programmes. It has provided some indication of the potential contamination profile of the facility during outages.

140 I am satisfied that Westinghouse has identified and quantified the radiological hazards associated with its plant and identified areas for improvement in order to restrict workers doses so far as is reasonably practicable, including reducing the source term or incorporating localised shielding. It is understood that a future licensee is likely to adopt its own scheme for area designation, but from the perspective of assessing the generic design of the AP1000, Westinghouse's zoning scheme is adequate and has been appropriately applied.

4.2.3 Normal Operation - Designated Areas - Findings

141 It is not clear whether entry to Containment will be required whilst at power and so the following Assessment Finding has been raised:

AF-AP1000-RP-02: *The licensee shall identify, and provide a justification for, all reasonably foreseeable work activities that are likely to require entry to the Containment whilst at power. For each of these activities, the licensee shall justify the reasons for this exposure, assess likely worker doses, and substantiate whether doses have been reduced so far as is reasonably practicable*.

This finding shall be addressed before fuel on-site.

142 In some cases, it is difficult to relate the radiation zoning of each area to the type of work which may be undertaken in that area and so it is not clear whether all reasonably practicable measures have been incorporated into the design in order to minimise external radiation levels. As a result, the following Assessment Finding has been raised:

AF-AP1000-RP-03: *The licensee shall, taking into account any changes to the radiation source term and shielding design which have been made since GDA, provide a report that identifies external dose rates for all controlled areas during normal operation. In addition, the licensee shall submit an ALARP justification for areas to which access is required and where the dose rate exceeds 150 micro-sieverts per hour (during normal operation) to demonstrate that dose rates have been reduced so far as is reasonably practicable. The justification shall consider the nature of the work which is likely to be conducted within those areas, including the magnitude and duration of the exposure, and the number of workers exposed.*

This finding shall be addressed before fuel on-site.

4.3 Normal Operation – Shielding

143 The utilisation of effective shielding is a key control measure for restricting the exposure of personnel and the public. It is a passive engineering measure that follows the minimisation of radiation source terms in the hierarchy of control measures (Ref. 19). As a result, I have considered shielding design to be a principal aspect of my assessment.

144 The shielding assessment was undertaken to assess the AP1000 shielding provisions identified in the PCSR submission (Ref. 12), to review the arguments presented in the PCSR, and to assess whether the evidence presented substantiated those arguments for shielding. The objectives of the shielding assessment were as follows.

- To be satisfied that the AP1000 shielding design fulfilled the requirements outlined in the SAPs (Reg. 4), in particular RP.6 and in the TAG for radiation shielding (Ref. 14).
- To be satisfied that relevant good practice had been applied to the shielding provisions to help to demonstrate that external dose rates and dose accrual by workers and members of the public were ALARP.

4.3.1 Normal Operation – Shielding - Assessment

145 In the UK, there is no specific legislation governing the requirements and acceptability of shielding provisions for facilities and so this assessment was carried out taking into

account international guidance from the IAEA (Ref. 51), the SAPs (Ref. 4), the TAG on radiation shielding (Ref. 14).

146 I raised a number of TQs (Ref. 8) to request information / documentation on shielding that was broad in content to provide an overview of information and a framework from which to undertake the specialist assessment. This TQ included a range of topics as indicated below:

- The adequacy of permanent shielding to protect the public during normal operation.
- The adequacy of permanent shielding to protect workers during normal operation.
- The adequacy of permanent shielding for enclosures required for the manipulation of radioactive substances.
- Arrangements for temporary shielding to protect workers from direct radiation during maintenance work, etc.
- The adequacy of materials used for shielding.
- The adequacy of shielding calculations methods and computer codes.
- The impact of long-term neutron activation on potential doses to workers during normal operation and decommissioning work.

147 The topic of shielding was assessed by ND's TSC, Nuclear Technologies (Ref. 40). The information and references provided in response to TQs were used as the groundwork for the assessment. The general areas considered within the shielding assessment were as follows:

- Design Criteria: This considers the design criteria used to assess the acceptability of shielding provisions.
- Shielding Design Basis Data: Review of source terms and physical data used as the basis for all shielding analysis.
- Radiological Classification of Areas: Consideration of the radiological zoning for each area of the nuclear island to understand the basic requirements for shielding provisions with regards to limiting external dose rates.
- Calculation Methods: Review of calculation methods, computational codes and their adequacy for use in shielding assessments.
- Shielding Assessments: General review of the adequacy of shielding provisions protecting members of the public and personnel.
- Dose Uptake: Consideration of how shielding assessments have been used in the prediction and restriction of dose uptake to levels that are ALARP.

4.3.1.1 Design Criteria

148 The design criteria used by the TSC were the dose limits and ALARP requirements identified in IRR99 and NT.1 Targets 1, 2, and 3 of the SAPs for normal operation,. These criteria are discussed in detail in Section 2.2, Standards and Criteria.

4.3.1.2 Shielding Design Basis Data

4.3.1.2.1 Shielding Source Terms

149 Shielding design is dependant on accurate yet suitably conservative radiation source terms. Shielding source terms were reviewed to ensure that the most significant sources of radiation with respect to reactor design (as outlined in Ref. 51) have been taken into account and are derived in a sufficiently conservative manner for shielding calculations and dose rate predictions (Ref. 14). All of the source terms that should be used for shielding assessments in support of the AP1000 design are presented in the Radiation Analysis Design Manual (Ref. 45).

150 The source terms for the reactor core, primary coolant and for spent fuel are the result of an extensive set of calculations using fuel specifications (i.e. initial enrichment, burn-up, power rating) combined with conservative factors to account for fuel failure (0.25%) and uncertainties in power output (Ref. 45).

151 For shutdown source terms, consideration has been given to the transport of primary circuit corrosion particulates to the core and subsequent activation and re-deposition throughout the primary coolant circuit. Conservative assumptions regarding deposition rates in ion exchange beds and filters have been used.

4.3.1.2.1.1 Containment Building at Full Power Operation

152 During full power operation, the dose rates and shielding requirements within the Containment building near the primary circuit are driven by oxygen activation products nitrogen-16 (gamma emitter) and nitrogen-17 (neutron and gamma emitter) within the primary coolant, in addition to the reactor core (neutron and gamma emission).

153 A detailed sample assessment was conducted on the generation of the neutron and gamma source terms at the boundary of the reactor cavity. This review was discussed in Section 4.1.1.1.

4.3.1.2.1.2 Containment Building during Shutdown

154 External dose rates within the Containment Building during shutdown will be dominated by gamma radiation emitters in the primary coolant such as activated corrosion products (cobalt-58 & 60 for example) and fission products as a result of fuel defects, if these should occur. Contributions to dose rates from neutron radiation arising from the core and neutron emitting activation products (e.g. nitrogen-17) will be negligible during shutdown given the short half-life of the most significant activation products (e.g. 4.2 seconds and 7.1 seconds for nitrogen-17 and nitrogen-16 respectively) (Ref. 45).

155 In addition to contributions to dose rates from the primary coolant in the circuit and in the pressure vessel, activation of structural steel, stainless steel components, concrete and neutron shielding material, due to neutron flux escaping from the pressure vessel into the reactor cavity, has been adequately considered.

156 Monte Carlo analysis of neutron transport in the reactor cavity provided neutron flux magnitude and spectra for input into a transmutation and activation code to determine activation source terms. This treatment was afforded to items such as the upper and lower neutron blocks, pressure vessel supports, hot and cold leg pipes and nozzles, nozzle gallery liner, pressure vessel lower shell, reactor vessel insulation system and the reactor cavity wall and liner. Activation of corrosion products in the primary coolant circuit

is treated separately from structural material activation. Account has been given to the production, transport, activation and deposition of corrosion products in the primary coolant circuit using the Westinghouse computer code CORA (Ref. 52).

4.3.1.2.1.3 Other Buildings of the Nuclear Island

157 The source terms used to calculate shielding provisions for other buildings of the nuclear island are solely based on gamma emitting nuclides as these will dominate dose rates. The gamma spectra of these sources are driven by activated corrosion products (e.g. cobalt-58 and cobalt-60) and by fission products arising from fuel failure (e.g. caesium-137). This is to be expected due to the nature of the radiation typically being either primary coolant or ion exchange resin based waste products and effluents.

158 Spent fuel source terms have been derived for a variety of decay intervals of between 12 hours and 5 years following shutdown. These are used when performing shielding calculations and potentially dose uptake assessments for spent fuel handling, storage and shipping. Spent fuel neutron source terms include contributions from spontaneous fission and alpha-n reactions with oxygen isotopes (oxygen-17 and oxygen-18). The neutron energy spectrum recommended in the Radiation Analysis Design Manual (Ref. 45) is the californium-252 spontaneous fission spectrum. When compared to other neutron spectra that can be used for spent fuel shielding calculations (e.g. induced uranium-235 and plutonium-239 fission spectra), the californium-252 spontaneous fission spectrum is adequately conservative.

4.3.1.2.1.4 Source Term - Summary

159 The TSC concluded that Westinghouse’s approach to defining the shielding source term was adequate and aligned with good practice in the UK. Conservative assumptions have been used in its generation and it is probable that actual dose rates on an operational plant would be lower than predicted.

4.3.1.2.2 Shielding Materials

160 In order to perform shielding calculations, it is necessary to have knowledge of the compositions and densities of specific shielding materials employed, as they have a significant effect on the material’s shielding performance. According to the shielding summary documents (Ref. 48, 49, 53, and 54) the typical materials employed in shielding are standard concrete, steel, lead and composite neutron shielding materials. The following summarises a review of the shielding materials (with respect to TAG 002, Ref. 14) as outlined in References 46, 48, 49, and 53-56.

161 The majority of the bulk shielding within the buildings encompassing the nuclear island is provided by the concrete walls and floor slabs, which provide both neutron and gamma shielding. The neutron shielding performance is largely driven by the isotopic composition and, in particular, the hydrogen content. The hydrogen content reduces towards a minimum level over time as the concrete ‘dries out’. The ability of concrete to attenuate gamma radiation increases with the density of the concrete. I raised a TQ to obtain information on the intended concrete composition and the composition and density of concrete provided by Westinghouse is considered appropriate for conservative shielding calculations where dose rates are driven by fission neutron and decay gamma radiation. The TSC also noted that steel rebar within concrete walls and floors (which is omitted

from shielding calculations) can often help to reduce dose rates further; this ensures that projected dose rates remain conservative.

162 The design of the AP1000 is such that access to the containment building, whilst at power, is minimised and so optimisation of neutron shielding within the containment building with the aim of reducing dose to operators has not been implemented in detail. Exceptions are the upper and lower neutron blocks fitted between the Reactor Cavity Liner and the Pressure Vessel in the Reactor Cavity. These have been fitted to:

- Limit neutron streaming upwards out of the reactor cavity at power to ensure that the dose rates on the Operating Deck and upper levels of the Containment Building allow a degree of man access at power.
- Limit neutron activation of the Nozzle Gallery and upper Pressure Vessel components.

163 As these shielding items are located close to the Pressure Vessel and are subject to extreme temperatures and radiation levels, traditional neutron shielding materials such as polythene or dense wood laminate would prove unsuitable. Westinghouse has identified a borosilicate resin as a shielding material and the TSC notes that this composition is similar to that of borosilicate neutron shielding resins, commonly found in transport flasks and is thus considered to be an appropriate compromise between shielding performance and thermal durability.

164 Shielding materials for gamma radiation have been identified as steel and lead, but the exact composition of these materials is not specified by the TSC as being overly significant when considering fission product gamma energies.

165 Appropriate account has been taken for the use of water shielding during refuelling and in areas such as the SFP. Variation of water density, temperature and pressure has been considered within various sections of the primary coolant circuit and in the generation of the Reactor Cavity boundary source term (Ref. 46).

166 Based on the evidence provided and the results of an in-depth review by the TSC, I conclude that the composition and densities of the shielding materials used in AP1000 shielding calculations are adequately conservative and consistent with those typically used in shielding assessments undertaken within the UK.

4.3.1.2.3 Flux to Dose Conversion Factors

167 Many radiation transport codes calculate particle flux, which is then converted to dose using energy dependant conversion factors. It was apparent that the regulatory standards in the US require the use of source data from different ICRP publications than those used in the UK, so I raised a TQ requesting information regarding the assumed neutron and gamma flux to dose conversion factors used in AP1000 shielding assessments (Ref. 8). The TQ response identified that these conversion factors come from:

- ICRP Publication 51 (Ref. 57) for point-kernel gamma ray calculations; and
- ANSI/ANS 6.1.1-1977 (Ref. 58) for Monte Carlo and discrete ordinates neutron and gamma calculations.

168 Although these are not the most recent flux to dose conversion factors recommended by ICRP Publication 74 (Ref. 59), a direct comparison of the gamma and neutron factors by the TSC demonstrated that in general, the factors assumed by Westinghouse are more conservative at lower energies for both neutron and gamma radiation. The gamma

response functions otherwise are not significantly different to the ICRP 74 response functions.

169 The differences between the neutron response functions vary by as much as 11% more conservative (at around 18-20 MeV) and 18% less conservative (at around 1.0-1.5 MeV). This could potentially result in an underestimation in contributions to neutron dose rates from a small portion of the neutron energy spectrum.

170 However, the consequences of this finding on actual dose uptake to personnel are expected to be negligible due to a number of mitigating factors as follows:

- Neutron contributions to dose uptake are only significant during access into the Containment Building at power. The allowance in the occupational dose evaluation document for man access at power is 100 hours per year (Ref. 50), which only represents a small portion of the total man hours per year for reactor operation.
- Neutron contributions to dose rate only arise from the RPV and from scatter from the internal Containment Vessel (CV). Ref. 11 shows that significant contributions from these sources (i.e. > 10%) only occur in locations with a Zone IV (1 $mSvh^{-1}$) classification or higher. Occupancy in these regions is expected to be much lower than the allocated 100 hours per year.
- In the locations where there are significant contributions to dose rates from the RPV, and from scatter from the CV, neutrons will only account for a small portion of the total dose rate (i.e. neutron plus gamma).

171 It is judged that the differences in response functions will not significantly affect the overall conclusions, with respect to shielding and dose rate estimates for AP1000 presented in Refs 12, 46, 48, 49, 53, 54, 55, 56, and 60.

4.3.1.3 Summary - Shielding Design Basis Data

172 I concur with the TSC's conclusions (Ref. 40) that the shielding design basis data is adequate and suitably conservative.

4.3.1.4 Radiological Classification of Areas

173 This part of my assessment considered the radiological zoning for each area of the nuclear island to understand the basic requirements for shielding provisions with regards to maximum external dose rates.

174 This topic is also discussed in Section 4.2., Designated Areas, of this assessment report and the shielding affects are included here for completeness to show that the radiological classification of areas is a significant component of the overall shielding assessment.

4.3.1.5 Calculation Methods

175 The shielding design of the AP1000 has employed a variety of calculation methods and computational codes. The choice of which methods and codes are used is generally dependant on the type of radiation and the complexity of the problem being assessed. The review carried out by the TSC considered the capabilities and adequacy of the codes and calculation methods used and whether they are appropriate for the specific shielding assessments to which they have been applied.

176 Brief descriptions of the shielding calculation methods employed in the design of the AP1000 are summarised in each of the shielding summary documents for buildings in the nuclear island (Refs 48, 49, 53, 54) and in the supplied detailed shielding assessments (Refs 46, 55 and 56).

177 Detailed neutron and gamma calculations of the shielding provisions for the Containment Building and some areas within the Fuel Handling Area of the Auxiliary Building have been undertaken using Monte Carlo or discrete ordinates radiation transport analysis.

178 For analysis of direct or bulk shielded gamma dose rates where the complexity of the problems at hand is somewhat reduced, more simple and less time intensive point kernel analysis has been employed.

4.3.1.5.1 Computational Codes

179 A variety of codes have been employed by Westinghouse in shielding assessments, with each being selected for a particular scenario based on the complexity of the model. Codes included MCNP5 (Ref. 61), DOORS 3.2 (package including ANISN, DORT and TORT) (Ref. 62), MicroShield™ (Ref. 63), VISIPLAN (Ref. 64), and SCAP (Ref. 65). These codes were judged by the TSC to be appropriate tools for conducting shielding assessments and were suitable for the scenarios to which they had been applied.

180 The TSC noted that it was unfamiliar with DORT and so undertook a cross-check of a dose rate calculation for refuelling operations using Attila (Ref. 66), a 3-dimensional discrete ordinates code, which is commonly utilised in shielding assessments within the UK. The finding was that the DORT and Attila results compared very well indeed, which provides confidence that the code performs as expected and that the logic followed by Westinghouse is similar to the approach taken by the shielding TSC.

181 The shielding TSC stated that, when used by suitably qualified and competent assessors, all of the above codes are considered to be capable of modelling the source data discussed in the Westinghouse Radiation Analysis Design Manual (Ref. 45).

4.3.1.5.2 Application of Codes in Shielding Assessments

4.3.1.5.2.1 Geometry Modelling

182 Based on the shielding documentation provided by Westinghouse, the shielding geometries used in calculations have been modelled in a conservative manner so as to ensure that the resulting dose rates will also be conservative. This often involves the omission of plant items from shielding model which could, in reality, provide some additional shielding benefit (for example, rebar in reinforced concrete has been excluded in prompt dose rate calculations).

4.3.1.5.2.2 Material Cross-section Data

183 The documentation provided by Westinghouse confirms that ENDFB-VI cross-section data has been utilised in calculations using the DOORS 3.2 package and using MCNP.

184 VISIPLAN (Ref. 31) does not use detailed cross-section libraries. Rather it relies on gamma ray attenuation and build-up factors from ANSI/ANS-6.4.3-1991 (Ref. 67).

MicroShield™ also uses Ref. 67 for attenuation coefficients and build-up data for many atomic elements.

185 These cross-section libraries are acceptable for fission neutron and decay gamma shielding assessment purposes.

186 No cross-section information was provided by Westinghouse with regards to calculations undertaken using SCAP.

4.3.1.5.2.3 Dose Points

187 According to Refs 48, 49, 53 and 54, dose points were generally placed at a distance of 1 foot (30cm) from the surface of the component or system from which the radiation arises, or from the surface of the shielding. Where the floor performs a shielding function, dose points were situated 3 feet (90cm) above the floor; approximately waist height.

188 Where simple calculations have been carried out in rooms with multiple sources, it was common practice for Westinghouse to take the calculated dose rate at 1 foot from each source and combine them to yield a conservative upper-bound estimate for the peak dose rate in the room. If this approach results in a breach of the radiation zone criterion for the room, the calculation may be revisited but typically the calculated dose rate will meet the zone criteria.

189 Where applicable, exceptions have been made to assess dose rates at the worst-case locations taking into account contributions from multiple radiation sources within the area and radiation streaming from adjacent areas via penetrations. This should ensure that the peak whole body dose rates to operators working within the room will meet the radiation zone criteria.

190 More detailed calculations make use of radiation contours created in MCNP to identify areas of high dose rate, for example on the Operating Deck in containment or for pipe penetrations in the wall separating the Normal Residual Heat Removal System (RNS) Valve Room from Steam Generator Compartment 2. The RNS Valve Room case is discussed specifically in Section 4.3.1.6.2.2.2. This practice is common in Refs 48 and 49 and is an effective way of identifying peak dose rates and dose rate trends, where results are close to the radiation zone criterion.

4.3.1.5.2.4 Result Accuracy

191 The statistical convergence for calculated dose rates quoted in documents provided by Westinghouse have all been within acceptable limits. For example, MCNP results quoted in the response to a TQ on the application of shielding assessments in dose optimisation have relative errors typically ranging from < 1% - 10%, which is acceptably within the <10% statistical error criteria outlined in Ref. 68.

192 There are instances where the statistical error exceeds 10%, sometimes by a considerable margin. However, this only occurs in positions where there are multiple calculations representing multiple contributing paths to operator positions. The dominant paths always demonstrate statistical errors of much less than 10%. Contributing paths with statistical errors greater than 10%, typically contribute several orders of magnitude less to the dose rate than the dominant paths.

193 The comparison between discrete ordinates codes discussed in Section 4.2.1.5.1 demonstrated that an increase in angular quadrature in the Attila calculation resulted in

no significant change in result. This indicates adequate convergence of the Attila calculation and therefore confirms that the DORT calculation had also adequately converged.

4.3.1.5.2.5 Cross-Checking

194 During a visit to Westinghouse's offices in Pittsburgh in August 2010 with ND assessors, the TSC discussed the subject of cross-checking of calculations in some detail. In general, the Westinghouse shielding team does not conduct cross-checks for every calculation. However, if results seem at all spurious to an assessor, it is common practice for the assessor to undertake a cross-check to verify their result. Alternatively, if the verifier of the calculations sees fit, they can undertake or request a further calculation by an independent method to provide confidence in the result.

4.3.1.5.2.6 Summary of Shielding Code Application

195 The documentation submitted by Westinghouse and confidence checks undertaken by the shielding TSC, generally demonstrates that:

- geometry modelling assumptions are suitably conservative;
- the cross-section data used for shielding calculations are adequate;
- suitable treatment has been afforded to dose point selection on a case by case basis;
- Monte Carlo results are quoted with an acceptable relative error and discrete ordinates calculations are suitably well converged; and
- although cross-checks are not carried out for every calculation; the quality assurance process provides a degree of confidence that should spurious results arise, adequate investigation and independent calculations will be undertaken.

4.3.1.6 Shielding Provisions

196 Within the scope of GDA, shielding summary reports, sample shielding assessments, plant layout drawings and responses to technical queries have been reviewed to better understand how the shielding design for the AP1000 has been developed. The following sections consider the adequacy of shielding provisions in the protection of both members of the public and personnel from direct radiation.

4.3.1.6.1 Protection of the Public from Direct Radiation

197 Protection of the public from external doses is achieved by adequately confining radiation to within the fabric of buildings within the nuclear island. This should also consider any potential radiation 'weak paths' due to penetrations (e.g. doors, vents etc).

198 A review of the bulk shielding provisions for the AP1000 has been performed to ensure that dose rates to the public are acceptable during all modes of normal operation (e.g. power operation and shutdown).

199 The bulk shielding provisions for the reactor shield building are assessed and provided in Ref. 56. This presents a detailed summary of the calculations undertaken to assess the shielding for the Containment Building.

200 The assessment considers:

- Containment Building bulk shielding – both primary reactor cavity shielding and secondary containment shielding;
- Passive Containment Cooling System (PCCS) air inlet penetrations; and
- PCCS air diffuser penetration.

201 These constitute both bulk shielded (direct) contributions and contributions due to scatter, streaming and weak paths to the dose rate external to the Containment Building. Furthermore, any contributions to dose rate from skyshine arising from the streaming of radiation through the high level PCCS penetrations have been considered.

202 The Reactor Cavity boundary source term document (Ref. 46) has been used for the Monte Carlo calculations considering direct and indirect (streaming/scattered) contributions from the reactor cavity at full power.

203 Contributions from nitrogen-16 in major primary coolant components (hot and cold legs, steam generators, pressuriser) were considered using point kernel calculations. Cold side contributions from these components were found to be negligible in comparison to those from the Reactor Cavity.

204 The results are presented in some detail. They confirm that dose rates outside the containment building are very small, typically of the order of 10^{-3} μSvh^{-1} at 50 metres from the shield wall. The results confirm that contributions to public dose due to external radiation can be expected to be negligible and will conform to the respective BSO.

4.3.1.6.2 Protection of Personnel from Direct Radiation

205 A review of the shielding provisions protecting personnel from direct radiation during normal operation has been undertaken. This initially considers the use of bulk shielding (such as shield walls, floors and doors) to ensure ambient whole body dose rates are in line with the dose rate design targets specified by the radiation zoning scheme. Further investigation into local shielding, penetrations and the use of temporary shielding has been undertaken on a sampling basis where it could have a significant impact on dose accrual.

206 It should be noted that certain local shielding items, such as gloveboxes and sample transport trolleys not associated with the generic reactor design, are a site-specific matter and therefore are not considered within the scope of the GDA shielding review.

4.3.1.6.2.1 Bulk Shielding Provisions

207 As discussed in Section 4.2.1.6.2 above, the bulk shielding provisions for the Containment Building will ensure dose rates at the exterior surface of the building will be acceptable.

208 There are three buildings of note in the nuclear island where bulk shielding will be required to protect personnel from ionising radiation. Furthermore, the Containment Building can be segregated into two different operational modes.

209 This results in a total of four separate radiation zoning schemes and sets of shielding assessments as follows:

- The Containment Building at power.

- The Containment Building during shutdown conditions.
- The Auxiliary Building.
- The Annex Building.

210 Although the detail and quality of these documents is very high, they do not necessarily present the results of final shielding analysis for the AP1000. Several documents recommend that further analysis is required in certain areas. They hence present detailed preliminary design shielding assessments with which a reviewer can gain confidence in the logic and methods employed. It is expected that the final shielding analysis will be completed only when the design is finalised at the site specific phase.

211 In general, shielding calculations have been undertaken using one or more of the following methods:

- Monte Carlo analysis using MCNP5 (Ref. 61):
  - for radiation streaming from the reactor cavity exploiting neutron and gamma source terms produced by Westinghouse in References 45 and 46;
  - for activated ex-core components using activation source term data from References 69 and 70; and
  - where more detailed treatment of gamma streaming/scattering than can be offered by Microshield™ / Visiplan / SCAP (Ref. 63-65).
- Point-kernel calculations using MicroShield (Ref. 63) for basic gamma dose rate calculations and using Visiplan (Ref. 64) for bulk shielding gamma dose rate evaluation in complex geometries with multiple sources. SCAP (Ref. 65) was used for point-kernel calculations where gamma scatter was a consideration.

212 The review of the Auxiliary Building shielding assessment was conducted first as this was the first substantial shielding assessment to be received by the shielding TSC. Comments raised during this review are generally applicable to all shielding assessments; therefore they have not been raised again when reviewing other buildings in the nuclear island.

4.3.1.6.2.1.1 Containment Building at Power

213 The radiation zoning scheme and the shielding assessments of rooms in the Containment Building whilst at power are presented in Ref. 48. This document considers calculated dose rates and compares them to the respective dose rate design targets specified by the zoning scheme.

214 The results show that all areas of the Containment Building whilst at power meet their respective criteria with the exception of:

- Room 11300: Maintenance Floor;
- Room 11303: Lower PRZ (Pressuriser) Compartment;
- Room 11304: SG1 Access Room;
- Room 11305: IRWST (In-Containment Refuelling Water Storage Tank); and
- Room 11504: Refuelling Cavity.

215 Radiation levels in the Lower Pressuriser Compartment, the East Steam Generator Access Room, the Refuelling Water Storage Tank and on parts of the Maintenance Floor are largely driven by the Surge Line in the Lower Pressuriser Compartment. It has been assumed that the Surge Line and Pressuriser source terms are evaluated assuming a 10% step load power decrease, resulting in an increased nitrogen-16 source term in the Surge Line and Pressuriser. Ref. 48 states that this assumption is overly conservative and recommends more refined analysis in order to yield less conservative dose rates.

216 Dose rates in other locations on the maintenance floor are primarily driven by the primary coolant legs in the Steam Generator 2 Compartment.

217 Ref. 48 demonstrates that the local steel shielding for the legs is insufficient and that increasing the thickness and span of this shielding could reduce dose rates on the Maintenance Floor considerably, albeit still not within the radiation zone criterion. Ref. 48 concludes that further analysis of alternative shielding solutions is required.

218 Dose rates in the Refuelling Cavity are driven by streaming from the Reactor Cavity. As such, dose rates at lower elevations meet their respective dose rate criteria but as dose rates increase with elevation, they no longer meet the criteria. In Ref. 48, Westinghouse recommends that either the radiation zoning scheme should be revised to better fit the actual dose rate profile or that the Refuelling Canal Gate be kept in place during operation as it could provide effective shielding.

219 The radiological protection considerations associated with recommendations made by Westinghouse have been captured in Assessment Findings **AF-AP1000-RP-01**, **AF-AP1000-RP-02** and **AF-AP1000-RP-03**, which have already been presented in this report.

4.3.1.6.2.1.2 Containment Building during Shutdown Conditions

220 The radiation zoning scheme and the shielding assessments of rooms in the Containment Building during shutdown are presented in Ref. 49. This document considers calculated dose rates and compares them to the respective dose rate design targets designated by the zoning scheme.

221 Contributions to dose rates from neutrons during shutdown are expected to be negligible. Percentage contributions to dose rates from Reactor Cavity Streaming and from Primary Coolant Circuit corrosion products have been documented by Westinghouse.

222 In general, calculated dose rates meet, or are close to, their respective design criteria. The following cases are notable exceptions:

(1) Maintenance Floor outside East Steam Generator Access Room (570 μSvh^{-1} in Zone II).

(2) Nozzle Gallery (41 $mSvh^{-1}$ in Zone V).

223 In both cases, Westinghouse shows that dose rates could be reduced by reducing cobalt impurities in stainless steel. This may result in bringing (2) down to within the Zone V criterion but will not be sufficient to bring (1) down to within the Zone II criterion.

224 Furthermore, the equivalent location for the North Steam Generator Access Room carries a Zone III classification, which raises some confusion with regards to whether the classification matches the occupancy expectations for these locations and what the expected radiation levels should be.

225 Most calculated dose rates have been demonstrated to meet their respective dose criteria, often by a large margin, but these dose rates remain high in some areas where maintenance and inspection is expected, e.g. the Steam Generator Compartments. Ref. 41 discusses comparisons with measured dose rates on current operating plants in Germany and concludes that AP1000 dose rates should be lower than those calculated and at least comparable with these plants.

226 There is some sensitivity analysis of cobalt content in key stainless steel components presented in the shielding assessment, and, as discussed in previous sections, relatively high cobalt stainless steel has been assumed for the majority of primary circuit components.

227 Dose rate contributions, due to corrosion products and activated stainless steel components around the Pressure Vessel in the Reactor Cavity, are roughly proportional to cobalt content in the steel.

228 Therefore the cobalt content of all components, which are in contact with primary coolant, should be reduced as far as is reasonably practicable. This would significantly reduce dose rates. This matter is the subject of Assessment Finding **AF-AP1000-RP-01** as discussed in Section 4.1.

229 Calculated dose rates in the PXS and RNS Valve Rooms are several orders of magnitude below the Zone VI design criterion. Ref. 49 states that other local sources that could contribute to dose rates at 24 hours following shutdown have been excluded. The reason for this is unclear but Westinghouse recommends further analysis in order to complete this part of the shielding assessment. This matter is the subject of Assessment Finding **AF-AP1000-RP-03** (as discussed in Section 4.2.1.1.4) and so I would expect it to be addressed during the site specific phase.

4.3.1.6.2.1.3 Auxiliary Building

230 The Auxiliary Building Shielding Assessment (Ref. 53) is a substantial document that considers dose rates in each of the rooms/areas in the Auxiliary Building.

231 There were a number of observations made by the TSC regarding this document, but the observations can be arranged into the following groups:

- It appears that some sources may have been excluded from shielding assessments of the rooms in which they reside. However, these tend to be high dose rate rooms and it is likely that their inclusion would not significantly alter the dose rates in the room and/or breach the zone radiation classification.
- Some dose rates breach their respective zone design targets with little or no justification as to why they are acceptable, i.e. with no statement why the dose rate and any associated dose uptake can be considered ALARP.
- Dual classifications are not always allocated where it may be deemed appropriate. Conversely, dual classifications are allocated to some areas where they could be deemed to be inappropriate. In general, there is a concern that the Radiological Classification of Areas may not always be applied to shielding assessments in the same way as would be expected in the UK.
- There are no recommendations made with regards to operational procedures where temporarily elevated dose rates may be present. Operational procedures may be a site specific, rather than a GDA matter, but recommendations should be made by

Westinghouse to ensure that the license holding utilities are aware of locations and instances where special attention may be required.

232 After the shielding TSC noted these observations, they were raised as part of a technical query. Westinghouse provided detailed responses to these observations, which are summarised henceforth (Ref. 8).

233 Concerning the observations made regarding the potential exclusion of important sources, satisfactory explanations for the source assumptions have been provided.

234 Westinghouse stated that, where necessary, revisions to the calculations and source assumptions will be included in the subsequent revision of the Auxiliary Building Shielding Assessment. It is standard practice for shielding assessments to be reviewed and updated repeatedly before the point of construction as part of the development of a shielding design based on individual operator working practices and preferences.

235 Westinghouse stated that the radiation zone classifications outlined in Ref. 53 are consistent with the Design Control Document (DCD) (Ref. 60) and are based on present plant design.

236 Shielding calculations have been undertaken, based on the most recent plant layout with regards to source and shielding configurations, to determine the suitability of the radiation zoning scheme. In summary, the purpose of Ref. 53 was not to undertake a detailed shielding design analysis but rather to identify areas where the radiation zoning scheme was unsuitable, or where calculated dose rates differ from the criterion by a significant margin. Where required, Westinghouse has stated that changes can be made through a Design Change Proposal (DCP) (Ref. 8) as the shielding design is finalised.

237 Westinghouse states that it will bring to the attention of licensees any issues that remain pertinent to the radiation zoning scheme and the dose rates calculated in the Auxiliary Building Shielding Assessment, for instance temporarily elevated dose rates in localised areas due to crud burst. This may be in the form of notes on the DCD drawings (Ref. 60) or in some other suitable format. Where Westinghouse feels that these issues warrant special attention from the licensee, it states that recommendations on operational controls (e.g. access restrictions) will be made (Ref. 8).

4.3.1.6.2.1.4 Annex Building

238 The Annex Building Shielding Assessment considers dose rates in each of the rooms/areas in the Annex Building. The large majority of rooms in the Annex Building either contain no source inventory, or have no neighbouring rooms that contain a significant source inventory.

239 For rooms that do contain significant source inventories, shielding calculations were undertaken to ensure that room classifications are met for both the host room and for neighbouring rooms.

240 Although some observations were made regarding the Annex Building Shielding Assessment, these are encompassed by the technical queries raised for the Auxiliary Building Shielding Assessment therefore there are no further observations to report.

4.3.1.6.2.2 Penetration Assessments

241 Westinghouse state that to minimise radiation streaming through wall penetrations, where practicable, offsets are incorporated between the radioactive source and the normally accessible areas. If offsets are not practicable, penetrations are located as far as practicable, above the floor elevation to reduce radiation exposure to personnel. If these two methods are not used, alternate means, such as baffle shield walls or grouting the penetration annulus, are used (Ref. 71).

242 No queries were raised regarding the treatment of penetrations. Several examples of assessments where penetrations/weaknesses in the bulk shielding require consideration are present in the supplied documentation. The following two examples demonstrate the treatment of two different penetration assessments.

4.3.1.6.2.2.1 Passive Containment Cooling System (PCCS) Air Inlets

243 Ref. 56 considers the PCCS air inlets on the Containment Building which potentially present a weakness in the Containment Building bulk shielding.

244 Radiation from the Reactor Cavity and primary coolant loop components could potentially propagate through the PCCS inlets. The penetrations are angled in such a way to eliminate direct streaming, i.e. only scattered radiation can propagate though the vents.

245 The system was modelled using the Monte Carlo code MCNP (Ref. 61) with appropriate variance reduction techniques to obtain results that pass the required statistical criteria. A separate calculation considered contributions through the surrounding bulk shielding with the PCCS inlets modelled as black body absorbers. This allowed Westinghouse to publish separate streaming and bulk shielded contributions to the external dose rate.

246 The results show that although the radiation emerging from the PCCS inlets is greater than that passing through the bulk shielding, the resulting potential dose to the public remains very small, as discussed in Section 4.2.1.6.1.

4.3.1.6.2.2.2 Pipe Penetrations in the RNS Valve Room

247 Ref. 48 considers pipe penetrations in the wall separating Steam Generator Compartment 2 from the RNS Valve Room. The penetrations have conservatively been assumed to be empty, i.e. it has been assumed that no fluids or cables that could offer any shielding benefit exist in the penetrations.

248 The results show that the streaming of radiation through the penetrations into the RNS Valve Room area results in a very localised region of elevated dose rate. However, even immediately in front of the penetration, the radiation zone criterion for the room is achieved. Given the localised nature of the elevated dose rate, the expected low occupancy at power and the fact that the zone criterion is observed, Westinghouse befittingly makes no recommendations for additional analysis.

4.3.1.6.2.3 Temporary Shielding Provisions

249 In general, it is understood that the use of temporary shielding in the operation of the plant is primarily a site specific matter and is therefore considered to be outside the scope

of GDA. However, it is felt that the following issues regarding Temporary Shielding require consideration as part of GDA.

250 A technical query was raised regarding whether the use of temporary shielding was implicitly included within the Occupational Dose Evaluation assessment (Ref. 50) as a result of using current operating plant data. Furthermore, it was queried whether any consideration had been given to provisions for utilising temporary shielding on the AP1000 facility.

251 If any dose savings due to temporary shielding have been implicitly incorporated into the Occupational Dose Evaluation, there should be adequate provision for the storage and use of temporary shielding incorporated into the AP1000 design.

252 In its response, Westinghouse states that the exposure from operating plant data used for the Occupational Dose Evaluation does reflect reductions offered by temporary shielding, although WEC view this as a “best practice” procedure and the use of temporary shielding is plant specific (Ref. 50).

253 Given this statement, it is important that although permanent shielding has been utilised as far as possible on AP1000, provisions for the use and adequate storage of temporary shielding should be available. Westinghouse has provided examples where such matters have been considered:

- Key components; pumps, coolers etc, have been designed to allow sufficient space for the installation of temporary shielding for maintenance.
- Embedment plates and lugs have been fitted to areas in the Auxiliary Building which can support temporary shielding if required.

254 The response to this technical query indicates that, although some account for the use of temporary shielding has been taken in the Occupational Dose Evaluation, permanent shielding has been utilised rather than temporary shielding where possible to reduce doses. Provisions for the use and storage of temporary shielding have been considered around key high dose items.

4.3.1.6.2.4 Summary of Shielding Provisions

255 The approach to the shielding assessments and the methods employed by Westinghouse are consistent with shielding practices in the UK. Where the TSC has identified discrepancies or areas for potential improvement, the RP has also recognised them and has subsequently made appropriate recommendations as discussed in the respective sections above.

256 It is accepted that the shielding assessment documentation submitted for review by Westinghouse is a “first pass” at the AP1000 shielding design in order to identify issues such as those observed by the shielding TSC. Accordingly, Westinghouse has made recommendations where further analysis will be required in subsequent revisions of the shielding assessment documentation. A formal process for design changes exists so revisions to shielding assessments can be documented. Where other observations have been made by the shielding TSC and raised as TQs, they have been responded to satisfactorily. Given this information, I consider that the submitted shielding substantiation documents are sufficient for GDA, and that the further refinements of the shielding provision will be progressed during the site specific phase as standard practice as the design is developed and finalised.

257 Therefore in summary, the available documentation supporting the shielding design for the Containment building at power demonstrates:

- Consistency with UK shielding design logic, methods and practice.
- That adequate recommendations have been made for further analysis where dose rates fail to meet design criteria.
- Consideration has been afforded to temporary shielding provisions to ensure that actual occupational dose uptake can be consistent with the Occupational Dose Evaluation assessment (Ref. 50), e.g. Embedment plates and lugs fitted where the use of temporary shielding is anticipated (Ref. 8).

4.3.1.7 Shielding Calculations in Dose Assessments

258 It is common practice in the UK for radiation dose uptake assessments to be conducted by shielding assessors using calculated dose rate data in order to generate conservative upper bound estimates.

259 Furthermore, the dose uptake model generated can be used to identify key contributors to the overall dose uptake assessment, either as a result of prolonged occupancy, elevated radiation levels or a product of both. This means that shielding assessments can be revisited with a view to reducing dose uptake from these key contributors as far as is reasonably practicable.

260 This subsequently helps to support the assertion that shielding design for a facility can be demonstrated to be ALARP as a whole.

261 A technical query was raised regarding this application of shielding assessment data to dose uptake optimisation. The response received from Westinghouse (Ref. 8) provided excerpts of detailed calculations that had been carried out to assess complex source/shield systems and where multiple dose points had been considered in order to provide input to dose optimisation studies. The examples included the:

- Demineraliser/Filter room (Room 12151) in the Auxiliary Building;
- Radwaste Building;
- Effluent Hold-up Tank Rooms (Rooms 12171 and 12172) in the Auxiliary Building;
- Delay and Guard Beds (Rooms 12153 and 12155) in the Auxiliary Building; and
- Spent Fuel Pit.

262 This response demonstrates that Westinghouse incorporated shielding calculations into occupational dose uptake estimates where either operational plant data is unavailable (e.g. for tasks not currently performed on operational plants) or for particularly complex shielding arrangements.

263 Such dose uptake estimates may be used for ALARP analysis of individual plant items even if they do not feed into the plant-wide occupational dose uptake assessment, which is based on actual operational plant data.

264 I am satisfied with Westinghouse’s approach of utilising a combination of operating plant data and calculated dose uptake models in order to substantiate occupational radiation exposure on the AP1000. However, the operational plant data must be applied only where the AP1000 design is comparable with existing operating plant. The fact that Westinghouse has identified areas where the operating plant data is not applicable and

has carried out specific dose uptake calculations in order to supplement this data, gives confidence that its assessment has been carried out competently.

4.3.2 Normal Operation - Shielding - Conclusions

265 Westinghouse has submitted all of the shielding summary documents for the Containment Building (both at power and 24 hours after shutdown), the Auxiliary Building, and the Annex Building, in addition to the UK PCSR submission Chapter 24 (Ref. 12). Further documentation has been issued in response to technical queries raised that included shielding design basis documents (i.e. the Radiation Analysis Design Manual, Ref. 45) and detailed calculation reports (i.e. Reactor Cavity boundary source calculations and Containment Building bulk shielding).

266 Overall, the standard of documentation received in support of GDA has been adequate to allow a sufficiently detailed examination of the AP1000 bulk shielding design and aspects of detailed shielding provisions.

267 All of the documentation submitted by Westinghouse and made available to the shielding TSC at the time of review, along with the responses to the technical queries directed to Westinghouse, demonstrate that the shielding design is being developed through a logical and iterative design process using acceptable methods, shielding codes and adequately conservative assumptions. The documentation demonstrates that the shielding is generally adequate to reduce dose rates to within the classification of areas criteria. Where this is not the case, Westinghouse has provided reasonable responses either justifying the breach of criteria or recommending further analysis. Assessment Finding **AF-AP1000-RP-03** has captured the requirement on a licensee to submit an ALARP justification that demonstrates that external dose rates in all accessible areas have been reduced for normal operation so far as is reasonably practicable (see Section 4.2.3).

268 The documentation submitted by Westinghouse supporting the AP1000 shielding design demonstrates that, when reviewed in the context of the guidance and expectations outlined in the SAPs (Ref. 4) and TAG (Ref. 14) the shielding provisions are acceptable. As a result, I have identified no reason why the shielding design of the AP1000 will not be capable of reducing external dose rates so far as is reasonably practicable.

4.3.3 Normal Operation - Shielding - Findings

269 There are no specific assessment findings for this topic.

4.4 Normal Operation - Optimisation for Work Activities

270 This part of my assessment focussed on the following aspects as detailed in my Step 4 assessment Plant (Ref. 1):

- Application of ALARP to all work activities.
- Prioritisation of ALARP for work activities involving the highest doses.
- Remote handling, remote observation, and use of robotics.
- Provision of radiological protection facilities.
- Collective doses and average doses for work activities.

- Substantiation of improvements leading to reductions in estimated doses.
- Management of doses during minor incidents.
- Doses to employees not working with ionising radiations.

4.4.1 Normal Operation – Optimisation for Work Activities – Assessment

271 Time at risk is covered in NT.2 of the SAPs (Ref. 4) which explain that there should be sufficient control of radiological hazards at all times. Guidance on time at risk is in Paras 629 to 638 of the SAPs (Ref. 4) and ND's TAG on the demonstration of ALARP (Ref. 32). Time at risk is geared mainly towards time at risk of the plant. However, for radiological protection, time at risk relates to time of exposure of the individual and guidance is provided on dose / risk sharing in the ACOP and guidance to IRR99 (Refs 19 and 22) and ND's TAG on the demonstration of ALARP (Ref. 32).

272 BSLs and BSOs for any workers in normal operation are covered in NT-1 Targets 1 and 2 of the SAPs and in Paras 585 to 589 (Ref. 4). General guidance on radiological analysis in normal operation (almost all of which is relevant to persons on the site) is provided in Paras 4.1 to 4.13 of T/AST/043 (Ref. 17). More specific guidance on radiological analysis for persons on the site (NT.1 Target 1) is in Paras 4.15 to 4.19, and for groups on the site (NT.1 Target 2) is in Paras 4.20 to 4.23 (Ref. 17).

273 Para. 4.22 of Ref. 17 explains that, although high dose work activities should have been analysed and the need for engineered provision included in the design, there may be tasks that could give rise to relatively high doses to specific workers and there should be a satisfactory ALARP assessment for these relatively high dose work activities. In addition, Para. 4.23 of Ref. 17 explains that future operators would be required to have adequate arrangements for assessing the average dose to specific classes of persons.

4.4.1.1 Performance Against Dose Limits and Targets

4.4.1.1.1 Westinghouse's Dose Assessment Methodology

274 The methodology used to calculate the AP1000 annual collective worker dose is specified in Chapter 24 of the PCSR (Ref. 12). The dose has been derived from two components:

- Historical dose data obtained from the operation of Westinghouse-designed 2-loop reactors, corrected to account for AP1000-specific design features, modern design improvements and improvements in operational practices.
- Data obtained using Job Exposure Model (JEM) calculations.

275 The baseline annual collective worker dose calculated for the AP1000 was 1680 Person-mSv and this had been developed from the collective dose calculated for the AP600 design, which was based on data from Westinghouse 2-loop reactors. This dose had been broken down into the following tasks:

- Reactor operations and surveillance.
- Routine maintenance.
- In Service Inspection (ISI).
- Special maintenance.
- Waste processing.

- Refuelling.

276 Westinghouse stated that the initial annual collective dose did not take account of improvements in the design and operation of its plants; namely:

- Reducing source term by reducing crud production and deposition:
  - Low cobalt tubing;
  - Zinc injection.
- Reducing component numbers:
  - Valves;
  - Pumps and heat exchangers (HX);
  - Demineralisers;
  - Filters.
- Specific design features relating to:
  - Refuelling operations;
  - SG Eddy Current (EC) inspection and tube plugging;
  - RCPs.

277 As a result, the AP1000 Annual Occupational Dose Evaluation (Ref. 50) was completed by Westinghouse in order to provide a less conservative estimate of collective dose. This value was then amended utilising data from the Prairie Island reactors in the US which had demonstrated continuous improvement and applying factors for reductions in the number of components and amount of crud.

278 Major differences in design between pumps and the reactor head and upper internals on the AP1000 design compared with previous reactors and the use of robotics for many operations, meant that it was not appropriate to scale doses from the following:

- RCP inspection.
- Refuelling.
- SG sludge lancing.
- SG secondary-side inspection.
- SG EC tube inspection.
- SG ISI.

279 The collective dose to the group of workers undertaking these tasks has been estimated using the JEM system, which uses a Microsoft Excel spreadsheet to estimate worker doses. The dose calculation is based on dose rates at existing plants, modified as appropriate for the reduction in crud as a result of the design and operating regime of the AP1000 plant and other design changes and durations of tasks based on the preliminary AP1000 plant refuelling outage schedule.

280 Using this approach, Westinghouse has estimated the total dose in a year in which an outage takes place as 239 person-mSv. This value includes the whole of the dose from the highest ISI (the RV, head, and SG ISIs each incur significant collective dose but are

not all carried out during the same outage, therefore, only the highest contribution has been included).

281 I consider the Westinghouse approach of utilising dose data obtained from operating plant to be appropriate because it often provides a more realistic estimate than can be obtained from theoretical calculation alone (which often involves inherently conservative assumptions). By supplementing this data with specific dose assessments for tasks which are unique to the AP1000, Westinghouse has demonstrated a logical and thorough approach to undertaking its dose assessments.

282 One concern associated with the approach used by Westinghouse involves the relevance and applicability of the data from other PWRs to the AP1000 design. If the reactors that were being used to obtain dose information differed significantly from the AP1000 in their design or planned operating regime for the UK, then the data extracted from them would be unsuitable for deriving a dose estimate for the AP1000.

283 In order to address this concern I raised a TQ (Ref. 8) to request Westinghouse to provide:

i) a list of the nuclear power plants whose dose/dose rate data were used in dose evaluations;

ii) a statement substantiating why these plants provide a reasonable estimate of doses and/or dose rates, including confirmation that:

   a) the data is applicable to the AP1000 design;

   b) the selected data is realistic (i.e. neither excessively optimistic nor excessively conservative).

iii) details of any efforts to compare the data with European PWRs.

284 In its response (Ref. 8) Westinghouse provided a list of reactors that had been directly utilised in the occupational dose analyses. Although most of these sites were based in the US, the list also included reactors from Sweden and Germany and also Sizewell B in the UK. In order to demonstrate that European operational experience had been taken into account, Westinghouse also provided details of two other programmes which had involved the exchange of relevant information:

- The “Four-Party Agreement” is a programme in which Westinghouse and three French organisations (EdF, Framatome and CEA) collaborated in data exchange/evaluation efforts. This programme extended over a period of approximately 10 years and provided a forum for technical information exchange between radiation experts from each of the organisations. Information on corrosion product input from SGs was shared in this programme.

- Westinghouse developed the EPRI-PWR Standard Radiation Monitoring Program (SRMP) in 1978. This program is still in-place and involves establishing a set of radiation monitoring points at approximately 10 standard locations in each of the primary coolant loops; including the SG and the hot and cold leg piping. Such measurements provide radiation field data at consistent locations and times after shutdown and have been used extensively in evaluating radiation fields from operating plants. The data is used in defining radiation trends for any particular plant and allows comparison of plant data between plants. The information from this program has been utilised in defining effective dose-reduction measures/techniques for Westinghouse and the industry.

285 With regard to the applicability of dose data obtained from operational PWRs, Westinghouse has stated that the major systems, components, and materials of construction associated with the AP1000 plant design are not radically different from those of the operating fleet of Westinghouse PWR plants (Ref. 8). Westinghouse claims that, in general, the major changes to the AP1000 from previous plant designs are in the area of accident mitigation and control, rather than normal plant operations; adding that these accident-related features generally entail fewer components, which lead to lower personnel exposure, since fewer components need to be maintained and repaired. In this sense, Westinghouse claims that these projections can be considered somewhat conservative.

286 TÜV SÜD carried out a review of the approach used by Westinghouse to estimate the AP1000 annual occupational collective dose. It came to the following conclusions (Ref. 41):

- The power plants used in the analyses are appropriate, having a good operating record in terms of radiological protection.
- The correction factors used by Westinghouse to derive AP1000-specific doses are appropriate and TÜV SÜD did not recognise any missing factors with significant impact on the dose estimate.
- TÜV SÜD did not identify any tasks which had been omitted from the dose estimate (the exception is doses incurred as a result of transferring spent fuel from the SFP to storage casks; this topic is addressed in Section 4.4.1.3.3.4).

287 One aspect of Westinghouse's submission which did cause concern was the focus on external radiation doses for workers, whereas doses from internal radiation sources resulting from inhalation and/or ingestion of radionuclides had not been assessed. As a result, I raised an RO to request that Westinghouse carry out an assessment of internal doses for the AP1000 under normal operation and to demonstrate that potential internal doses had been reduced so far as is reasonably practicable (Ref. 9).

288 Westinghouse's response outlined the summary of a study it had conducted of radioactive intakes at US reactor sites. This information was gathered from Nuclear Regulatory Commission (NRC) reports.

289 The response stated that personnel received internal doses at 55 currently operating units between 2006 and 2008. The following statistics were derived from the NRC reports (Ref. 9):

- Approximately 58% of internal dose events involved 5 or less personnel.
- Approximately 85% of internal dose events involved 15 or less personnel.
- Approximately 85% of the average internal doses per personnel were 0.25 mSv or less.
- Approximately 65% of the combined totals for internal doses for all personnel were 0.5 mSv or less per plant.
- 80% of the total internal doses were less than 0.1% of the Total Effective Dose Equivalent (TEDE) (assuming that TEDEs for plants that report multiple units together are evenly distributed between the units).

- For plants in which the TEDE represented a single unit, approximately 77% of the Committed Effective Dose Equivalent (CEDE) values were less than 0.1% of the TEDE.

290 Several of the intakes reported were incurred as a result of mechanical failure or operator error. This data suggests that any internal doses from normal operations are likely to be small when compared to external doses. However, it should be noted that this consideration does not diminish the duty on a potential future licensee to consider internal radiation doses in any site specific safety cases.

291 In summary, I am satisfied that the dose assessment methodology used by Westinghouse to estimate annual occupational collective doses is adequate, and covers all significant work activities.

4.4.1.1.2 Performance Against Numerical Targets in SAPs – Normal Operation

292 The SAPs provide dose targets for assessment of new facilities against regulatory requirements and relevant good practice. These targets include the Basic Safety Level (BSL), which is sometimes also the legal limit, and the Basic Safety Objective (BSO). Westinghouse has compared its dose estimates to these numerical targets and Westinghouse's conclusions on the outcome of this analysis is detailed in Chapter 24 of the PCSR and summarised in this section. My assessment of Westinghouse's claims, arguments and evidence with regard to ALARP are provided in Section 4.4.1.3 of this report.

Table 4: SAPs NT.1 Target

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 (IRR).*	

293 Westinghouse states that the highest individual dose from a single operation is provided in the Refuelling Dose Estimate (Ref. 72) as 2.02 mSv from disconnection and connection operations on the In-core Instrumentation System (IIS) at the Reactor Vessel Closure Head (RVCH). Although this dose provides some confidence that the BSL will not be challenged, this dose is above the BSO and is received in a short time (1.6 hours). Westinghouse claims that it is possible for two workers to undertake this task so that the individual dose is approximately 1 mSv (it is unlikely that more than two workers could be involved). It is likely that some workers will receive exposures from other operations during refuelling and so some individual doses may exceed the BSO and require ALARP justification. Since it is not possible to identify all of the tasks that workers will undertake

over the course of an entire year because the deployment of workers will be based on the preferences of licensees, it has not been possible to accurately assess the maximum annual individual worker dose. However, the magnitude of the doses presented by Westinghouse for individual tasks provides confidence that exposures to individual workers can be restricted to levels, which are significantly below the BSL.

294 Although the PCSR explicitly states that the highest individual dose from a single operation is 2.02 mSv, information in an appendix to Chapter 24 of the PCSR (Ref. 12) contradicts this by providing worker doses associated with the inspection of the Reactor Vessel Head. Westinghouse states that the highest individual exposure is 5 mSv, resulting from the removal of the inspection equipment following J-weld examinations, taking 0.5 hr at 10 $mSvh^{-1}$; although it does note that actual doses are expected to be lower. The magnitude of this dose and the short duration of the exposure is a significant concern. However, this dose is not included in the summary table in the same chapter, nor is it repeated within the main body of text in Chapter 24. The discovery of this inconsistency occurred at a late stage of Step 4 of GDA and so I have not undertaken an assessment of this activity and it remains unclear as to whether this dose is realistic, or whether it is the result of an error whilst compiling the information in the PCSR from the referenced calculation notes. However, if the predicted dose is accurate, then a future licensee will be expected to review this exposure as part of the response to Assessment Finding **AF-AP1000-RP-03**, which requires the justification of all dose rates exceeding 150 μSvh^{-1} in areas where access is likely to be required.

295 Areas with dose rates of <2.5 μSvh^{-1} are designated as Zone I areas. Westinghouse claim that, although it is possible that employees who do not work with ionising radiation could receive a dose of 5 mSv for an occupancy in these areas of 2000 hours per year, it is highly unlikely that they will reach this dose because most Zone I areas will have doses below 2.5 μSvh^{-1} and those areas with detectable radiation levels at power are low occupancy (Ref. 12). Dose rates in areas with higher occupancy, such as offices in the Annex Building, should be at or near background radiation levels (Ref. 12).

296 Two examples of areas outside the nuclear island where dose rates are locally elevated are provided in the Chapter 24 of the PCSR, but these areas are intended to have low occupancies. The dose rates are judged to be conservative and transient in nature and so should not result in significant exposures (Ref. 12).

297 Westinghouse states that there is no reason to believe that radiation from AP1000 plant operations will result in any employee in a non-radiation area receiving additional radiation exposure above background radiation levels or exceeding the BSO (Ref. 12). I have seen no evidence to suggest that this claim is inaccurate.

Table 5: SAPs NT.1 Target 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	

298 Westinghouse claims that the average dose to all radiation workers on-site depends on the total number of workers but is expected to be less than 1 mSv. This claim presumes that a minimum of 240 personnel will work on an AP1000, which appears to be a reasonable assertion. The doses to specific groups of workers also depend on the number in each group and the dose assigned to the group.

299 The group identified by Westinghouse as potentially having the highest average exposure, is the group of 32 workers undertaking the refuelling operation. The average dose to this group is estimated to be 1.3 mSv (Ref. 12).

Table 6: SAPs NT.1 Target 3

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 (IRR).*	

300 Regulation of public radiation exposure is shared between the Environment Agency (in England and Wales) and HSE. ND leads for HSE on doses to the public resulting from direct radiation (i.e. direct radiation originating from within the site boundary) during normal operation. The Environment Agency leads on doses to the public resulting from discharges of radioactive waste into the environment during normal operation, and so this topic area is outside the scope of my assessment report.

301 Westinghouse asserts that the primary shielding provided by the reactor and the concrete primary shield; the secondary shielding around the SGs, pressuriser and other primary circuit components; and the shield building around the containment ensure that dose rates outside during operation are less than 2.5 μSvh^{-1} based on conservative estimates of the source term. Westinghouse's calculations indicate that the dose rate at 35 m from the Shield Building is 0.0081 μSvh^{-1}, 0.0035 μSvh^{-1} at 90 m and 0.0011 μSvh^{-1} at 170 m. The dose rate profile has a maximum of 0.0085 μSvh^{-1} at 50 m as a result of scattered radiation through the passive containment cooling system air vents in the shield building (Ref. 12).

302 The total annual predicted dose, including both external radiation and representative discharges is 14 µSv, so it is clear that doses from external radiation will be below the BSO. The reduction of public exposure so far as is reasonably practicable, by the use of shielding was discussed in Section 4.3.1.6.1.

303 As discussed in the AP1000 Step 3 Assessment Report for radiological protection (Ref. 6), the collective annual dose, which has been presented for the AP1000, is in the order of that reported by the best-performing PWRs which are currently operating throughout the world, including the 'Konvoi' reactors, which I consider to represent relevant good practice with regard to worker doses.

304 Westinghouse has undertaken a detailed campaign of estimating worker doses and these comply with the BSL doses for targets 1 and 2 (for any person and any group on the site). Although the BSOs are exceeded, the magnitude by which they have been exceeded is not excessive and conservatisms in the source terms should mean that actual doses

would be lower. Furthermore, public doses resulting from direct radiation are likely to be negligible and certainly below the BSO. The application of ALARP at the design stage is considered in the following sections.

4.4.1.2 Application of ALARP

4.4.1.2.1 Westinghouse's Approach

305 In addition to reducing the source term associated with the AP1000 design and utilising shielding to attenuate direct radiation, Westinghouse also outlines its approach to minimising occupational radiation exposure in the AP1000 ALARA Guidelines Manual (Ref. 71). The matters covered in the document are discussed hereafter and can be grouped under the following headings:

- Reducing working time;
  - Elimination of tasks;
  - Minimising maintenance and inspections;
  - Improving work efficiency;
  - Minimising the number of workers exposed;
- Maximising distances between sources and workers;
  - Robotics;
  - Remote working;
  - Plant layout.
- Control of contamination;
  - Containment;
  - Housekeeping;
  - Ventilation.

4.4.1.2.1.1 Reducing Working Time

306 Westinghouse claims to have incorporated features into the design, which eliminate the need for undertaking certain tasks where possible. It describes the permanent reactor cavity seal ring as an example of this approach (Ref. 72). The permanent seal ring eliminates the need for temporary inflatable rubber seals, which must be installed and removed during each refuelling and sometimes require refitting due to leaks. Critical path time is lost in reinstalling the seal ring and the radiation doses incurred can be substantial. Westinghouse suggests that the installation of the seal ring at one plant led to a dose of 360 person-mSv (Ref. 72).

307 Westinghouse briefly discusses the reduction in the number of components associated with the AP1000 relative to older generations of PWRs in the main body of Chapter 24 of the PCSR (Ref. 12). The Annual Occupational Dose Evaluation (Ref. 50) describes component reduction factors, reproduced in Table 7 below, which represent the ratio of the number of components in the AP1000 design to the number in the previous Westinghouse reactor plants:

Table 7: AP1000 Component Reduction Factors

Component Type	Component Reduction Factor
All valves	0.40
Manual valves	0.37
Safety class valves	0.61
Pumps	0.43
Heat Exchangers	0.56
Demineralisers	0.63
Filters	0.63

308 As discussed in Section 4.4.1.1, these component reduction factors were used to adjust the existing dose data from operational reactors in order to estimate doses for the AP1000. Although I have not performed a detailed assessment of the suitability of these component reduction factors and the associated reductions in operator exposure, I have seen no evidence to suggest that Westinghouse's claims are inaccurate. As a result, I am satisfied that these values illustrate that reducing the numbers of components in the design has led to a reduction in maintenance and inspection requirements, with the result of reducing occupational exposures.

309 Westinghouse states that Systems, Structures, and Components (SSC) are designed for reliability to reduce the need for breakdown maintenance and maintainability to reduce the duration of maintenance operations. It claims that this is demonstrated by the ability of Westinghouse plants to achieve a high level of availability. The average availability of Westinghouse design reactors was over 91 percent between 2000 and 2008 and a number of plants achieved an availability of 100 percent for years where there is no refuelling outage (Ref. 12). It suggests that design improvements made to the SGs over the years provides an example of increased reliability and reduced inspection requirements. Another example is the use of multiple electric lights, so that the failure of a single light will not necessitate entry for replacement.

310 Westinghouse claims to have made simple modifications which optimise work efficiency and so minimise worker doses, such as improved lighting in the work area, or improved tools and procedures for specific tasks. The reactor vessel stud tensioning/detensioning procedure is given as an example of personnel dose reduction resulting from a change in procedures and a reduction in time. A review of historical procedures allowed Westinghouse to eliminate a significant number of steps involved with the operation and it claims that this resulted in estimated time savings of approximately 16 person-hours, with significant associated dose savings (Ref. 72).

311 Westinghouse also claims that the use of a motor-driven stud spin-out tool, allows rapid removal and installation of reactor vessel studs and its use minimises time spent in a high radiation area (Ref. 72).

312 The AP1000 design includes a facility named the ALARA Briefing Room, where preplanning activities, including mock-up training, layout familiarisation with photographs,

detailed tooling specifications and any check points or precautions, result in an enhanced ability to perform the work quickly and efficiently in the radiation environment. The provision of this facility represents good practice.

313 Westinghouse claims that the SG primary manway cover handling fixture is an example of a design feature which will minimise the number of workers exposed to radiation as part of the removal and reinstallation of the 318 kg manway cover. It states that the fixture will permit two people to safely and reliably remove a manway cover, while four to six workers would be required without the use of the fixture (Ref. 72). As a result, collective doses for this task will be reduced.

314 The adoption of the measures outlined above will depend on decisions made by future licensees. It is clear that Westinghouse has considered design features in order to enable their use and so, from the claims and evidence provided, I judge that Westinghouse has demonstrated sufficient efforts to reduce operator working times.

4.4.1.2.1.2 Maximising distances between sources and workers

315 Westinghouse claims that remote/robotic techniques have been developed for repetitive operations that would result in high operator doses where the additional dose incurred during setting up and removing the equipment still results in a worthwhile net dose saving. The following robotic/remote systems have been accounted for in the routine dose uptake assessment (Ref. 12):

- Eddy current inspections of SGs and tube, eliminating the requirement for worker entry into the SG head.
- Examination of the RVCH and penetration welds, eliminating the need for worker access beneath the head when on its storage stand.

316 ND conducted a visit to the Westinghouse research and development facility and assessors were shown examples of robotic technology which could perform these tasks. It has been demonstrated that the AP1000 design accommodates the use of this equipment and that its application represents relevant good practice.

317 Westinghouse has identified additional tasks where doses can be reduced by maximising distances between sources of radiation and workers, such as removing pumps to low dose rate areas for servicing and utilising reach rods to operate valves. It estimates that the use of the latter tool could result in an annual dose reduction of up to 0.52 person-mSv (Ref. 72). Remote observation techniques such as closed-circuit television (CCTV) and fibrescopes, are also identified as being useful for in-service inspections of highly radioactive components, in addition to underwater television systems for remote viewing of refuelling operations and core mapping functions. It is understood that the application of these tools will be decided by licensees.

318 Westinghouse has considered the following facility design features directed toward minimising radiation levels in plant access areas and near equipment requiring personnel attention (Ref. 12) as outlined below:

- Radiation sources are separated from occupied areas, where practicable (for example, pipes or ducts containing potentially highly radioactive fluids do not pass through occupied areas).

- In those systems where process equipment is a major radiation source, the pumps, valves and instruments are separated from the process component to allow servicing and maintenance of these items in reduced-radiation zones.
- Redundant components requiring periodic maintenance that are a source of radiation, are located in separate compartments to allow maintenance of one component while the other component is in operation.
- Control panels are located in low-radiation zones.
- Shielding is provided to separate equipment such as demineralisers and filters from non-radioactive equipment to provide unrestricted maintenance of the nonradioactive equipment. Labyrinth shields or shielding doors are generally provided for compartments from which radiation could stream or scatter to access areas and exceed the radiation zone dose limits for those areas.
- For potentially high radiation components (such as ion exchangers, filters, and spent resin tanks), shielded compartments with hatch openings or removable shield walls are used.
- Equipment in non-radioactive systems that requires lubrication, is located in low-radiation zones.
- For radioactive systems, adequate space and ease of movement in a properly shielded inspection area are emphasised. Where longer times for routine inspection are required and permanent shielding is not feasible, space is provided for portable shielding.
- Wherever practicable, lubrication of equipment in high-radiation areas is achieved with the use of tube-type extensions to reduce exposure during maintenance.

319 I consider that these measures are appropriate and represent relevant good practice with regard to the layout of nuclear facilities.

4.4.1.2.1.3 Control of Contamination

320 This topic is discussed in Section 4.5.

4.4.1.3 Demonstration of ALARP for High Dose Tasks

321 In order to assess the claims, arguments and evidence described in Westinghouse's submission, a number of tasks were selected for sampling. This process was assisted by TÜV SÜD, which compared the practices against similar activities undertaken at PWRs in Germany.

4.4.1.3.1 Refuelling

322 The principal document, which describes operational exposure resulting from refuelling activities, is the AP1000 Radiation Exposure Estimate for Refuelling (Ref. 72). This document describes the individual tasks involved with the refuelling programme, the maximum dose rates for working areas, the anticipated number of workers exposed and the duration of exposure. The dose rates utilised were obtained from existing plant; corrected for improvements in the AP1000 design. Using the JEM programme, the

aforementioned data has been used to calculate individual doses and total collective doses for a refuelling campaign.

323 The total collective dose is given as 43.8 person-mSv, with the majority of the dose being incurred during the Reactor disassembly and Reactor assembly phases, with a combined collective dose of 37.7 person-mSv. I noted that this calculation has been revisited prior to its inclusion in the appendix in Chapter 24 of the PCSR (Ref. 12), leading to a reduced annual collective dose of 41.3 person-mSv. It appears that this is the result of a reduction in the number of workers required for certain tasks.

324 TÜV SÜD compared the total collective dose to doses reported at German PWRs (Ref. 41). It found that, when compared against the average collective dose of eleven PWRs, the AP1000 figure was lower, with the German reactors averaging approximately 70 person-mSv. However, when comparing against three 'Konvoi' plants, the AP1000 collective dose estimate was approximately double that of the German plants. It concludes that the AP1000 doses are in the order of those reported for the third generation, or pre-'Konvoi' reactors.

325 TÜV SÜD notes that the dose rates used in the calculation should be conservative and the assumption that workers occupy these maximum dose rates for the full task is also conservative. Although a detailed assessment of the ergonomics and working positions involved with refuelling has not been undertaken, it suggests that actual dose rates should be lower than those presented (Ref. 41).

326 The work areas involved in the calculation are designated as Zones I to VI, with dose rates ranging from 10 μSvh^{-1} to 7.5 $mSvh^{-1}$. The vast majority of work is carried out on the Operating Deck, which is a Zone I area and, as might be expected, the number of person-hours of occupancy in each area decreases as its designation increases, with Zone VI areas having a total occupancy of less than 10 minutes (Ref. 12).

327 As stated in Section 4.4.1.2, the maximum individual dose during refuelling is given in Refs 12 and 72 as 2.02 mSv for the disconnection of the IIS bullet-nose assemblies. Westinghouse claims that this exposure has been minimised by the use of quick-lock devices on the bullet-nose assemblies, which significantly reduce disconnection and connection times when compared to the use of conoseal clamps (Ref. 12). This remains a high dose for a short duration exposure and Westinghouse has suggested that the task could be shared between two workers (four in total, since this is a two person task). The Approved Code of Practice (ACOP) for IRR99 Regulation 8, advises radiation employers to give priority to improving engineering controls and adopting other means of restricting exposure over and above dose sharing between employees (Ref. 22). I am satisfied that Westinghouse has incorporated engineering control measures in order to reduce dose but I have not investigated whether there are any further reasonably practicable measures that could be adopted to achieve further reductions. However, Assessment Findings **AF-AP1000-RP-01** (minimisation of cobalt content in primary circuit components) and **AF-AP1000-RP-03** (justification of external dose rates in accessible areas) may also further restrict the exposures received by these individuals. A future licensee would also be likely to reduce doses using administrative arrangements such as effective pre-planning and mock-up training.

328 In addition to the above control measure, Westinghouse has claimed the following design features for demonstrating that refuelling doses have been reduced so far as is reasonably practicable (Ref. 72):

- Integrated Head Package (IHP) which eliminates a separate missile shield and a number of steps in the plant refuelling. The IHP design includes shielding provisions (i.e. 2 inch thick steel shroud) above the reactor vessel head and a cable bridge design that accommodates electrical disconnects at the operating deck with quarter-turn bayonet-style electrical connectors rather than from above the top of the CRDMs.

- CRDM Cooling Fans Cam Type Disconnects that eliminate the need to access nuts and bolts on all sides of the cooler fan plenum. The fans are supported on the CA40 module.
- Quick Opening Fuel Transfer Tube Closure System replaces a conventional bolted cover. The associated time savings is important to exposure reduction since crud tends to accumulate in this area.
- Smooth Finish Reactor Cavity Liner reduces the contamination level in the work area and the time spent in decontamination of the cavity. It also reduces the requirements for respiratory protection and protective clothing.
- Permanent Reactor Cavity Seal Ring rather than bolted or inflatable types of cavity seals that must be installed and removed at each refuelling.
- Permanent Guide Studs which eliminate the guide stud installation and removal operations.
- Quick Release Reactor Head Insulation with “suitcase-type” fasteners and permanent ID marking, rather than insulation fastened with screws and with no permanent markings.

329 In summary, Westinghouse has undertaken a detailed assessment of occupational doses associated with refuelling, including both collective and individual doses and has demonstrated that it has used its experience of operating plants to incorporate design features into the AP1000 which will reduce doses. Based on the evidence presented, I am satisfied that Westinghouse has demonstrated that anticipated worker doses associated with the operation of an AP1000 have been reduced so far as is reasonably practicable. As required by IRR99, a detailed ALARP assessment will be required at the site specific stage which accounts for the licensee’s own working practices.

4.4.1.3.2 Steam Generator Inspections and Maintenance

330 At my request the TSC undertook a review of two tasks on the SGs; eddy current tube inspection and tube plugging, and sludge lancing.

331 Westinghouse’s calculation for eddy current tube inspection and tube plugging presented a collective dose of 7.5 person-mSv (Ref. 12), based on one third of the tubes being inspected, three tubes being plugged, and two SGs being inspected (the collective dose for one SG inspection is 3.7 person-mSv).

332 Westinghouse’s calculation for sludge lancing presented a collective dose of 12.8 person-mSv (Ref. 12), based on the sludge lancing of two SGs (the collective dose for one SG inspection is 6.4 person-mSv). This activity constitutes the largest contribution to collective dose for routine maintenance.

333 Westinghouse claims the following SG design features reduce doses to workers involved with these activities (Ref. 12):

- The tube ends are designed to be flush with the tube sheet in the SG channel head to eliminate a potential crud trap.

- The SG manways (entrance to channel head) are sized for easy entrance and exit of workers with protective clothing and to facilitate the installation and removal of tooling.
- The SG design includes a sludge control system/mud drum, designed to reduce the need for sludge lancing and reduces tube and tube support degradation.
- The design of SG tube support plates and the full-depth tube sheet expansion of tubes reduce corrosion and occupational exposure.
- The potential for using robotic technology for eddy current inspections of SGs and tube plugging, eliminating the requirement for worker entry into the SG head.

334 Westinghouse claims that dose rates for these EC inspection and tube plugging would be reduced by electropolishing the SG Bowl, but no credit was taken in the dose estimate. The minimum effect would be a reduction of 0.48 person-mSv per SG or 0.96 person-mSv total. It notes that there may be a reduction of some of the other dose rates where the SG Bowls make a contribution.

335 It is clear that Westinghouse has taken steps to incorporate design features into the SGs which will minimise the radiological risk to personnel, including allowing the possibility of using robotics as a remote technology, which represents relevant good practice. However, some dose rates associated with this work are significant and the extended exposure durations may lead to high collective doses (for instance, a residence time of 12 hours is required for sludge lancing in a dose rate of 150 μSvh^{-1}).

336 The TSC carried out a comparison of doses resulting from SG work on the AP1000 against those obtained at ‘Konvoi’ reactors in Germany. It reported that a typical collective dose for SG related work in one outage is below 5 person-mSv (Ref. 41). It also stated that the typical collective doses for SG inspection and SG lancing would be below 2 person-mSv per outage. The TSC expressed particular concern with the magnitude of the dose for sludge lancing, highlighting the fact that the task of moving and removing sludge lancing equipment from an SG would incur a collective dose of nearly 2 person-mSv, as a result of the average dose rate being 40 μSvh^{-1}, whereas the same activity in a ‘Konvoi’ plant would take place in a dose rate averaging less than 10 μSvh^{-1}.

337 It is understood that the projected AP1000 SG dose rates are likely to be conservative and that actual operational doses would be lower than those presented. Taking this into account, I judge that Westinghouse has reduced the radiation exposure of personnel undertaking EC inspection and tube plugging so far as is reasonably practicable. However, the doses incurred as a result of SG lancing are excessive when compared against relevant good practice at ‘Konvoi’ sites and Westinghouse has failed to adequately justify the enhanced doses. It is likely that reducing cobalt impurities from the primary circuit will reduce dose rates (see Section 4.1), but it is my opinion that the dose assessment for SG lancing will need to be revisited at the site specific phase in order to provide realistic worker doses and to demonstrate that occupational exposures from these activities have been reduced so far as is reasonably practicable. As a result, this topic is the subject of Assessment Finding **AF-AP1000-RP-04:**

AF-AP1000-RP-04: *The licensee shall provide a justification for doses incurred during SG lancing, taking into account changes to the radiation source term and shielding design which have been made since GDA. The report shall include an assessment of measures which will reduce worker doses further. This finding shall be addressed before fuel on-site.*

338 A possible contributing factor to the enhanced dose rates involved with SG work is the design feature of attaching the RCPs to the SG, leading to an increase in the source term at the SG manway due to direct radiation. Since this feature differentiates the AP1000 from previous generations of PWRs, I raised a TQ (Ref. 8) to request that Westinghouse justify that decision. Westinghouse responded with the claim that this design feature:

- Eliminates the SG to RCP suction “cross-over” piping, reducing the number of loop piping welds and eliminates the need for “cross-over” piping supports. The removal or reduction in the number of welds from the design reduces inspection requirements for these welds and associated doses.
- Eliminates the need for the RCP support structure and results in the ability to support each RCS loop, consisting of the Hot Leg (HL), Cold Leg (CL), RCPs, and SG, with one simple support below the SG channel head.
- Is a key feature in the AP1000 loop piping arrangement, which employs a single piece HL and single piece CL, further reducing the number of loop piping welds.
- Allows the AP1000 loop arrangement to be very compact and contributes to the ability to minimise the containment vessel diameter. This maximises the containment vessel design pressure capability.
- Results in improved RCP suction conditions (minimises the Net Positive Suction Head available to the RCP) and orients the pump such that the possibility of trapping air in the pump motor assembly is minimised and venting requirements are also minimised. Trapped air can lead to damage or failure of the motor assembly. A damaged or failed motor assembly would require additional repair or replacement operations, resulting in increased operator dose.

339 After considering this response, I concur with Westinghouse’s assertion that, although there is the possibility for the additional dose rate contribution from the RCPs at the SG manway, the benefits of placing the RCPs in this location outweigh the potential increase in dose. However, this consideration should not diminish Westinghouse’s efforts to further reduce doses incurred during SG related work, as discussed above.

340 Chapter 20, Appendix 20C of the PCSR (Ref. 12) states that the SGs have a component lifetime of 60 years and so are not expected to be removed during the normal operational life of the plant. As a result, I have not considered the radiological protection aspects associated with their removal. In the event that a future licensee wished to undertake this task, it would need to be managed as a substantial project requiring a safety case which considers the specific radiological conditions of the plant at the time of removal.

4.4.1.3.3 Other Activities Sampled for Assessment

4.4.1.3.3.1 Maintaining the Chemical and Volume Control System (CVS)

341 HSE ND’s Reactor Chemistry Assessors notified me of a potential line of enquiry with regard to the positioning of the AP1000 Ion Exchange Beds and this led to sampling of doses incurred while changing ion exchange resins.

342 The Ion Exchange Beds in the AP1000 design are located inside the containment, differentiating it from other PWRs, where the beds are positioned outside containment. It appeared that the AP1000 approach may increase the doses to personnel involved in the changing of Ion Exchange Beds and, as a result, a TQ was raised (Ref. 8) which asked Westinghouse to provide the following information:

i) justification of the positioning of the Ion Exchange Beds within containment;

ii) an overview of the tasks involved with changing the Ion Exchange Beds, including the radiological conditions at the point of work and on access routes, the number of workers involved, and the task duration;

iii) an estimate of anticipated worker doses from the changing of Ion Exchange Beds; and

iv) a description of any features included within the design which have been included to reduce doses so far as is reasonably practicable.

343 Westinghouse’s response claimed that its approach requires fewer valves, a reduction in the length of piping and, importantly, a reduction in the number of containment penetrations (Ref. 8). These features lead to decreased requirements for containment isolation tests, in-service testing and in-service inspections, with corresponding decreases in operator doses.

344 The AP1000 Occupational Radiation Exposure Estimate for the Chemical and Volume Control System (CVS) (Ref. 73) sets out the main tasks involved with maintaining and inspecting the CVS, detailing the rooms that operators occupy, the anticipated dose rates that will be encountered and the expected duration of exposure. It also recommends measures for reducing doses further, such as improving shielding.

345 This exposure estimate states that the Cation Bed Demineraliser is expected to be changed out once every three years while a Mixed Bed Demineraliser will be changed out once every 18 months. This averages out to one demineraliser changeout per year, for an annual operator dose of approximately 180 µSv. The annual worker dose for a change out whilst at power is given as 330 µSv during at-power conditions; an increase of approximately 150 µSv compared to the same operation during shutdown conditions (Ref. 73). Westinghouse states that in normal plant operation, it is expected that the demineraliser changeouts will be undertaken during shutdown conditions. It specifies that the at-power dose is only given as a reference and will not be included in the overall dose of the system.

346 The TQ response (Ref. 8) identifies several existing design features intended to reduce operator exposures, including shielding and Reach Rods in Room 11209 and remotely operated valves in the CVS and WSS.

347 The doses involved with resin changes and the control measures identified above, suggest that this activity is unlikely to be of significant regulatory concern when compared to other high dose activities undertaken during shutdown.

348 As a result of reviewing the document, an additional activity was highlighted which appears to involve significant exposure to radiation. The additional operator dose incurred for a Reactor Coolant Filter (RCF) changeout while at-power is given as 9.2 mSv when compared to the same operation being conducted during shutdown. For context, the total dose incurred from all operations and maintenance associated with the CVS while shutdown is given as 4.8 mSv per year (Ref. 12). Westinghouse states that the majority of the dose is obtained from the operation of a filter drain valve, which is exposed to radiation from the RPV and calculations indicate that 6 mSv could be incurred in five minutes (Ref. 73). Westinghouse indicated this as a concern within the report.

349 Further assessment has addressed this matter by clarifying that each RCF assembly is sized to operate for an entire fuel cycle without requiring a filter cartridge changeout (Ref. 81). As a result, Westinghouse states that exhausted filter cartridges will be changed

during plant shutdowns. Furthermore, Westinghouse claims that an automated device will be employed which is capable of de-tensioning the bolts on the filter vessel head, removing the head and exchanging the spent filter cartridge with a clean one. Afterwards, the device is capable of replacing the vessel head, re-tensioning all of the bolts and transporting the spent cartridge to the storage location in the Auxiliary Building (Ref. 81). All of these functions are conducted remotely in order to avoid unnecessary personnel exposure to the high radiation levels associated with this equipment. I have not undertaken an assessment of the evidence underpinning the claims associated with this equipment.

4.4.1.3.3.2 Inspection and Maintenance of Automatic Depressurisation System (ADS) Valves

350 Interactions with Westinghouse and the Mechanical Engineering assessors had suggested that certain aspects associated with positioning and design of the ADS valves, might be unique to the AP1000. This presented the potential that operator doses associated with the maintenance and inspection of these valves might not be accurately covered AP1000 Annual Occupational Dose Evaluation (Ref. 50) which had been based upon data from existing plant. As a result, a TQ was raised which requested that Westinghouse provide the following information:

- An overview of the tasks involved with the inspection and maintenance of Stage 1, 2 and 3 ADS valves, including the radiological conditions at the point of work, the number of workers involved, and the duration of the tasks.
- An estimate of anticipated worker doses incurred from the inspection and maintenance of Stage 1, 2 and 3 ADS valves.
- A description of any features included within the design which have been included to reduce doses so far as is reasonably practicable.

351 The response from Westinghouse claimed that the tasks associated with the maintenance and inspection of the Stage 1, 2 and 3 ADS valves, are similar to any other safety-related motor operated valve and that the only regular maintenance and inspection activities that will be associated with these valves will be the verification of the valve setup. This task involves attaching a test connector to the actuator of the valve. The test connector has a plug-in connection and therefore does not require any significant amount of time for setup. Once the test connector is attached, the valve will go through a series of tests and the results of these tests are recorded by an operator. Depending on the plant's requirements, this task will require either one or two operators. Altogether, the tests will take about 1 hour to complete per ADS valve.

352 In total, there are four Stage 1, four Stage 2, and four Stage 3 ADS valves, which are evenly split between Train A and Train B. The Train A valves are located in the upper tiered ADS valve room (Room 11703) while the Train B valves are located in the lower tiered ADS valve room (Room 11603). The projected dose rate at shutdown conditions near the lower tiered ADS valves is about 80 μSvh^{-1}, while the projected dose rate near the upper tiered ADS valves are about 8.3 μSvh^{-1}. Therefore, the maximum dose for the Train B valve setup verification would be about 160 person-μSv per valve while the Train A valve setup verifications would be about 16.6 person-μSv per valve. Using a conservative assumption of two valves from each train being tested per outage, Westinghouse claims that this would result in a total of 353.2 person-μSv.

353 The RP claims that there are several design features, which reduce the overall exposure that results from the maintenance and inspection of these valves:

- The valves themselves have been designed to extend the time between valve setup verifications as much as possible.
- The valves are also located at a distance from the pressuriser, as opposed to being seated directly on top of it, which it claims was the case on some older designs. This will result in lower dose rates at the valve locations.
- There is permanent scaffolding around the valve locations. This removes the requirement to construct temporary scaffolding, which reduces the overall time to complete the task.

354 Worker doses associated with the maintenance and inspection of ADS valves and the design features identified above, suggest that this activity is unlikely to be of significant regulatory concern when compared to other high dose activities undertaken during shutdown.

4.4.1.3.3.3 Radioactive Waste Processing

355 The collective annual dose for waste processing provided in the PCSR (Ref. 12) is 23.2 person-mSv for an outage year, with this dose being derived from recorded doses at US reactors and adjusted to account for improvements in design and operational techniques. The contribution from radioactive waste handling is given as 12.9 person-mSv.

356 An attempt is also made in the PCSR (Ref. 12) to derive a more accurate dose by utilising assumptions concerning area dose rates and occupancies. This approach results in a collective dose of 11.2 person-mSv per year. The largest proportion of this dose results from work in the Radwaste Building and is stated as 8 person-mSv per year, assuming an average dose rate of 6.5 μSvh^{-1} and a collective occupancy of 1280 person-hours for four workers. This approach is very simplistic and does not take into account hazards and ergonomic factors which are specific to each work activity.

357 In order to verify the accuracy of the radioactive waste processing figures, I raised a TQ (Ref. 8). This TQ noted that the estimated doses for waste processing presented in the AP1000 Annual Occupational Dose Assessment were calculated using operational data from nuclear power plants in the US. I stated that it is likely that waste processing at a prospective AP1000 facility in the UK would require a greater degree of contact with radioactive waste as a result of increased sorting and monitoring associated with UK requirements, with potentially higher doses being incurred by personnel. As a result, Westinghouse was asked to review the waste processing doses and to specify whether they provide a realistic estimate of those doses which are likely to be received by operators at UK plants.

358 Westinghouse’s response acknowledged that there may be differences in the way that the UK plants will handle radioactive waste compared to how it is handled by the US plants. It argued that the current radioactive waste management case for the UK plants is very generic and is only applicable to the GDA. It added that a more detailed radioactive waste management case and radioactive waste treatment facility design will not be completed until the site specific phase of the design. The design of radioactive waste facilities is discussed in the next section.

359 Westinghouse has made an attempt to quantify worker doses from radioactive waste processing, but it is my opinion that this is not suitable and sufficient for a reactor operating in the UK because it utilises data from US plants, where regulatory requirements for waste processing and so the practices which lead to worker exposure

from these activities, may differ. However, I accept that the design of radioactive waste facilities will be subject to change during the site specific phase and so it will be difficult to accurately estimate doses until these designs are finalised. As a result, I have captured this matter in Assessment Finding **AF-AP1000-RP-05**:

> ***AF-AP1000-RP-05:*** *The licensee shall carry out an assessment of realistic doses resulting from waste processing and provide a report which substantiates that these doses have been reduced so far as is reasonably practicable. This finding shall be addressed before the first fuel load.*

4.4.1.3.3.4 Spent Fuel Pool

360 The detailed assessment of worker doses resulting from the process of transferring spent fuel from the Spent Fuel Pool (SFP) to an on-site store is considered to be a matter for the site specific phase. However, information exchanged with Westinghouse on the adequacy of its criticality control safety case have referred to the doses of workers involved with the loading of dry storage casks (Ref. 9), with Westinghouse claiming that doses are influenced by the timing of this task and the number of loading operations per campaign. The magnitude of these doses is not quantified.

361 It is my opinion that the transfer of fuel from the SFP to on-site storage is an operation that can be planned in advance, so that engineering controls (e.g. shielding and remote working) are used as the principal control measure to restrict the exposure of workers. Provided that these engineering controls are robust, I would expect that doses from this activity should be low, but the information provided by Westinghouse (Ref. 9) has raised concerns that the hierarchy of control measures is not being adequately applied to this aspect of the design of the facility.

362 Considering the restrictions on space in some areas of the AP1000 discussed in Section 4.4.1.3.4 below, I have concerns that similar restrictions may apply to facilities involved with the loading and processing of spent fuel casks prior to their transfer to an on-site store. These potential restrictions might limit space for shielding or cause personnel to work in closer proximity to radiation sources than might be expected. As a result, I have raised an Assessment Finding (**AF-AP1000-RP-06**) that requires the licensee to substantiate that there is sufficient space in the Fuel Handling Area to load and process a transfer cask in order to allow the despatch of fuel to an on-site storage facility, while also restricting worker doses so far as is reasonably practicable.

> ***AF-AP1000-RP-06:*** *The licensee shall provide a report that demonstrates that there is sufficient space in the Fuel Handling Area to load and process a transfer cask in order to allow the despatch of fuel to its chosen design of on-site storage facility, while also restricting worker doses so far as is reasonably practicable. This finding shall be addressed before first structural concrete.*

4.4.1.3.4 Design of Radiological Protection Facilities

363 During an ND visit to the Westinghouse offices in Pittsburgh in February 2010, Westinghouse presented a series of layout diagrams for health physics and radioactive waste facilities on a generic AP1000 site (some of which are presented and described in the PCSR, Ref. 12). I expressed concern about the restricted amount of space, which had been allocated to these facilities and commented that it could challenge the radiological safety of personnel working within these areas. Along with the ND assessors

from the radioactive waste and Decommissioning topic area, I also had concerns that this lack of space would reduce the capacity of the design to safely manage radioactive wastes. As a result, a TQ was raised in order to obtain more supporting information with regard to the layout of these facilities and to allow a more detailed assessment of this issue (Ref. 8).

364 After reviewing the response to this TQ, our concerns were ongoing and, as a result, an RO was raised requesting that Westinghouse should demonstrate that the proposed health physics and radioactive waste facilities have been designed, so as to restrict the doses of personnel working within the facilities to so far as is reasonably practicable (Ref. 9). The response to that regulatory observation provided useful information which enabled ND to perform a more detailed assessment of facilities and the outcome was that we consider the design of the health physics and radioactive waste facilities to be inadequate, in that the amount of space allocated to them is insufficient.

365 In particular, our assessment of the design of the health physics and radioactive waste facilities identified several deficiencies; examples of which are provided below:

- According to the information provided, a worker who has been undertaking tasks within a potentially contaminated area in the controlled area would pass through one permanently installed contamination monitor upon exiting the controlled area and then change/wash in the change room. There are no additional permanent monitoring and changing facilities provided. This single layer of contamination control does not demonstrate defence in depth and could lead to a greater potential for contamination being carried outside the controlled area.
- In addition to the comment above, the amount of space allocated to the proposed change rooms (both male and female) was restrictive and, in my opinion, is unlikely to have the capacity to accommodate the number of workers requiring the use of these facilities during periods of high demand. This feature of the design could lead to workers bypassing the facilities (e.g. failing to wash), with a potential loss of contamination control.
- The size of the Primary Chemistry Laboratory is restrictive, with insufficient space for personnel to work safely. Radioactive materials are routinely handled in this area and the poor ergonomics could lead to an increased risk of accidents occurring. The amount of space allocated to other health physics radiometric work, including the HP Counting Room, also appears restrictive, leading to a potential failure of operators to efficiently process safety-related samples.
- The amount of space allocated to radioactive waste facilities is restrictive. Areas which are allocated for the receipt, processing, monitoring, packaging and storage of radioactive wastes are insufficient to allow radioactive materials to be managed safely and effectively. It is reasonably foreseeable that these facilities would be unable to deal with the volume of waste generated during times of increased demand, such as during and following outages.
- The amount of space allocated to the Hot Machine Shop is not adequate. There would be several activities with the potential to generate surface and airborne contamination taking place in close proximity to each other, creating the potential for unnecessary exposure being received by workers. As described earlier, there are no permanent changing or monitoring facilities installed in this area, so the potential for contamination to be carried outside the room is significant.

- The design does not accommodate a radioactive source store within the nuclear island, which would be utilised for the safe and secure storage of sources required for the routine calibration and function testing of radiological instrumentation, or for higher activity sources which might require temporary storage during site radiography campaigns.

366 It was noted, however, that there were examples of good practice, including the incorporation of an ALARA Briefing Room into the design. There were also significant improvements made to the layout of the solid radioactive waste processing areas in the RO response when compared to the original TQ response, which demonstrated efforts to segregate hazardous activities and improve contamination control.

367 My concerns were supported by my experience of UK nuclear facilities in addition to information gathered during visits by HSE ND to overseas NPPs. These sites had allocated significantly more space to these facilities than is detailed in the AP1000 submission. My views are also supported by an assessment of the facility carried out by ND's TSC, TÜV SÜD, which compared the designs to existing reactor facilities in Germany (Ref. 41).

368 In summary, I consider that Westinghouse has not adequately demonstrated that the AP1000 health physics and radioactive waste facilities have been designed so as to restrict the doses of personnel working within the facilities so far as is reasonably practicable. In particular, it is my view that insufficient space has been allocated to many of the health physics and radioactive waste facilities detailed in the response to the RO and that this may lead to an increased likelihood of radiological accidents and increased doses being incurred by workers. As a result, I do not consider the design of the facilities detailed in the response to the RO to be acceptable. However, a Westinghouse response to a TQ raised by the Environment Agency (Ref. 8) on multi-unit sites, outlined a potential strategy which involves using centralised facilities for health physics and solid radioactive waste management activities. Since these facilities are located outside the nuclear island, it should be possible for them to be constructed without the physical restrictions on space which are apparent in the proposed layouts detailed within the response to the RO.

369 Consequently, this matter can be addressed during the site specific phase, and therefore this topic is captured as Assessment Finding **AF-AP1000-RP-07**:

> ***AF-AP1000-RP-07:*** *The licensee shall ensure that suitable and sufficient space is available in the design and layout of the site specific health physics facilities (including laboratories, changing/monitoring facilities, emergency facilities and permanent decontamination facilities). The licensee shall provide a justification that the site specific design reduces worker doses, and reduces the likelihood and severity of reasonably foreseeable radiological accidents within the facility, so far as is reasonably practicable. The justification shall be supported by a suitable and sufficient human factors assessment. This finding shall be addressed before first structural concrete.*

4.4.2 Normal Operation – Optimisation for Work Activities - Conclusions

370 As discussed in the AP1000 Step 3 Assessment Report for radiological protection, the collective annual dose which has been presented for the AP1000 is in the order of that reported by the best-performing PWRs which are currently operating throughout the world, including the 'Konvoi' reactors which I consider to represent relevant good practice with regard to worker doses.

371 Westinghouse has undertaken a detailed campaign of estimating worker doses and these comply with the BSL doses for targets 1 and 2 (for any person and any group on the site). Although the BSOs are exceeded, the magnitude by which they are exceeded is not excessive and conservatisms in the source terms should mean that actual doses would be lower. Furthermore, public doses resulting from direct radiation are likely to be negligible and certainly below the BSL and BSO.

372 I am satisfied that Westinghouse has applied the hierarchy of control measures, incorporating engineering controls and other safety features, which have been derived from operational experience and should successfully reduce worker doses. In particular, Westinghouse has demonstrated that features that facilitate the use of robotic and remote technologies have been incorporated into the design. Furthermore, there is evidence that design improvements have been targeted at activities which have the highest worker doses, such as engineering controls which enable exposure times associated with refuelling to be decreased.

373 In summary, I am satisfied, based on the evidence presented, that Westinghouse has reduced the exposure of workers at an AP1000 reactor so far as is reasonably practicable. Nevertheless, there are some areas for improvement and Westinghouse has generally been forthcoming at identifying these during its own dose assessments. Those which I consider to be significant have been captured as assessment findings.

4.4.3 Normal Operation – Optimisation for Work Activities - Findings

374 Dose rates associated with SG lancing work appear not to be ALARP and so the following Assessment Finding has been raised:

> ***AF-AP1000-RP-04:*** *The licensee shall provide a justification for doses incurred during SG lancing, taking into account changes to the radiation source term and shielding design which have been made since GDA. The report shall include an assessment of measures which will reduce worker doses further. This finding shall be addressed before fuel on-site*.

375 The estimated doses submitted by Westinghouse for waste processing are not likely to include all tasks which would be undertaken at a UK nuclear site and so the following Assessment Finding has been raised:

> ***AF-AP1000-RP-05:*** *The licensee shall carry out an assessment of realistic doses resulting from waste processing and provide a report which substantiates that these doses have been reduced so far as is reasonably practicable. This finding shall be addressed before the first fuel load.*

376 There are concerns regarding the restriction of exposure to personnel who are undertaking the task of transferring spent fuel from the SFP to on-site storage and so the following Assessment Finding has been raised:

> ***AF-AP1000-RP-06:*** *The licensee shall provide a report that demonstrates that there is sufficient space in the Fuel Handling Area to load and process a transfer cask in order to allow the despatch of fuel to its chosen design of on-site storage facility, while also restricting worker doses so far as is reasonably practicable. This finding shall be addressed before first structural concrete*.

377 The design of health physics facilities is not adequate and so the following Assessment Finding has been raised:

> ***AF-AP1000-RP-07:*** *The licensee shall ensure that suitable and sufficient space is available in the design and layout of the site specific health physics facilities (including laboratories, changing/monitoring facilities, emergency facilities and permanent decontamination facilities). The licensee shall provide a justification that the site specific design reduces worker doses, and reduces the likelihood and severity of reasonably foreseeable radiological accidents within the facility, so far as is reasonably practicable. The justification shall be supported by a suitable and sufficient human factors assessment. This finding shall be addressed before first structural concrete.*

4.5 Normal Operation – Contaminated Areas

378 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of contaminated areas would include the following matters.

- Sources of contamination (e.g. primary circuit, fuel ponds).
- Minimisation of the generation of surface and airborne contamination.
- Application of the hierarchy of control measures to contamination.
- Monitoring of workplaces, articles and workers.
- Use of shielded containment (e.g. cells, glove boxes).

379 Contamination is a topic that spans both GDA and site specific phases since there are design features that can help to prevent and mitigate contamination, but the nature and extent of contamination is also dependent on the work activities and processes and procedures of future operators.

380 This topic is linked to Section 4.2 on designated areas, Section 4.8 on decontamination and health physics and radioactive waste facilities in Section 4.4.1.3.4 in this assessment report.

381 I described the approach of Westinghouse to contamination zoning in Section 4.2.1.2. Internal doses, which are closely related to contamination control arrangements, are discussed in Section 4.4.1.1.1.

4.5.1 Normal Operation – Contaminated Areas - Assessment

382 The radionuclides, which may be present in surface and airborne contamination, are addressed in the PCSR (Ref. 12) which describes the sources and likely exposure routes of radionuclides which may constitute a radiological risk. The majority of these are only likely to be encountered during maintenance activities. The measures employed by Westinghouse to reduce source terms, will also minimise the generation of contamination when sealed containment systems are opened.

383 Westinghouse claims that the AP1000 plant is designed to contain radioactive liquids and gases within vessels, pipes, and pumps. Potential sources of contamination, with the exception of the SFP, are isolated from normally occupied areas. Liquid sampling systems are designed to maintain containment (Ref. 12).

384 Since contamination control arrangements will be selected by a licensee during site specific assessments, my assessment of contamination control has focussed on the health physics facilities described in Westinghouse's submission. Westinghouse has provided arrangement diagrams for the following facilities:

- Change rooms and washing facilities.
- Contamination monitoring facilities.
- Decontamination facilities (described in Section 4.)

385 Westinghouse states that all personnel access to the RCA of the Annex Building, the Auxiliary building, the Shield building, the Containment and the Radwaste Building is via a single entry/exit area in the Annex Building and this is the location for permanent monitoring and changing facilities. Personnel are expected to change their clothes before entering the RCA, collecting additional PPE (e.g. respirators) from the PC Pickup and Suit-up Room as required. Access into the containment is via shielded personnel airlocks on the 100-m and 110.744-m operating deck level and via equipment shield doors on the same levels. When exiting, they must monitor themselves on full body monitors before changing and washing in the male or female change room (Ref. 12).

386 The PCSR (Ref. 12) states that all personal equipment and portable tools taken into the RCA must be monitored at the Health Physics Booth before removal from the RCA and decontaminated, retained in the area, or disposed of as active waste. Decontamination of tools can be undertaken in the Hot Machine Shop. Only essential items will be permitted to be taken into the active area to minimise the monitoring, decontamination, and waste generation.

387 My concerns about the design of the health physics facilities are outlined in Section 4.4.1.3.4. In particular, I judge that there is little evidence of defence-in-depth with regard to contamination control. Personnel leaving containment pass through one full body monitor before changing and washing. If this monitoring fails to detect contamination then there is a risk that personnel may carry contamination away from the Annex Building without their knowledge.

388 Westinghouse claims that local access controls will be implemented close to areas under maintenance with temporary barriers for additional PPE and monitoring as necessary. While this is common practice for areas on reactor sites where it is not possible to position permanent access controls, such as alongside plant items which are maintained on an infrequent basis, efforts should be made to ensure that facilities for changing, monitoring, washing, etc. are available for permanent work areas requiring frequent access. Referred to in the UK as sub-change rooms, these facilities are generally much smaller than the main change facilities but should act as a barrier to support contamination control at the point of work.

389 I have identified two areas that were lacking permanent sub-change rooms, where I considered the inclusion of such facilities to be necessary and reasonably practicable. These were the Hot Machine Shop and the Radwaste Building; both of which will be used for handling and processing unpackaged contaminated items and have relatively high levels of occupancy. Westinghouse has since amended its layout of the Radwaste Building to include a permanent sub change facility (Ref. 9). The installation of permanent contamination control facilities should also be considered for areas such as active laboratories.

390 Assessment Finding **AF-AP1000-RP-07** requires that improvements are made to health physics facilities and these improvements are likely to lead to a reduced risk of

contamination being transferred outside facilities such as the Hot Workshop. As a result, this Assessment Finding can be addressed by considering each facility in isolation. However, in order to ensure that there is effective defence-in-depth with regard to contamination controls on the site, it is important to understand the pathways which may lead to contamination being carried outside the controlled areas. An effective method of achieving this is by carrying out a human factors assessment which considers:

- The types of workers that may be exposed to contamination.
- The types of work activities which may lead to personnel being exposed to contamination.
- The potential transfer routes involved with a spread of contamination outside the controlled area.
- The engineering and administrative control measures which have been developed to prevent or minimise the potential spread of contamination.

391 A suitable and sufficient human factors assessment would be expected to address all of these factors and would be used as part of a wider risk assessment in order to substantiate the effectiveness of the contamination control arrangements.

392 Westinghouse has demonstrated that it has incorporated engineering controls into the AP1000 design, which will minimise the generation of contamination and also the potential for it spreading. It has also identified measures further down the hierarchy of contamination control measures, which can be utilised by future licensees. However, some omissions from the submission, such as the inclusion of permanent sub change rooms in potentially hazardous and high occupancy areas, have caused concern. As a result, the topic of contamination control arrangements, including 'defence in depth' is the subject of Assessment Finding **AF-AP1000-RP-08**:

> ***AF-AP1000-RP-08:*** *The licensee shall provide a justification that substantiates that defence in depth has been applied to the contamination control aspects of the AP1000 design, taking into account the response to AF-AP1000-RP-07. In particular, the report shall demonstrate that there are suitable and sufficient barriers to the spread of contamination from controlled areas to non-designated areas. The justification shall be supported by a suitable and sufficient human factors assessment. This finding shall be addressed before fuel on-site.*

393 The AP1000 design features specified in the PCSR (Ref. 12) for reducing airborne contamination (and as a result, internal doses) in working areas are listed as:

- improved SFS decontamination flow (maximising cleanup of particulate in the SFP) and cooling (minimising evaporative losses of the SFP);
- physical boundaries between radiologically controlled areas and non-radiologically controlled areas;
- the inclusion of an ALARA briefing room in the plant layout; and
- comprehensive ventilation of plant areas.

394 Westinghouse also claims that a comprehensive Radiation Monitoring System design is provided in order to provide information and warning notifications regarding contamination levels in order to reduce potential internal doses.

395 Taken in isolation, I consider these measures to be appropriate, but have not assessed airborne contamination control measures in detail. Ventilation and the Radiation Monitoring System are discussed in Sections 4.6 and 4.7 respectively.

396 Other arrangements for minimising and controlling airborne contamination, such as the use of tented enclosures around hazardous work areas, will be determined by future licensees. I have not assessed the availability of space for erecting these enclosures.

397 One particular area of concern regarding the exposure of personnel to both surface and airborne contamination, is the Fuel Handling Area. The PCSR (Ref. 12) states that the area is classified as amber (surface contamination possibly >4 $Bqcm^{-2}$ and airborne activity <0.1 DAC) and there is the potential for contamination to be present in the SFP water, which may become airborne. SFP water is constantly filtered by the SFS, but the reactor chemistry assessors have raised concerns that activated corrosion products within the pool water may be higher than anticipated during certain periods of operation, and that this could create a radiological protection risk to personnel (Ref. 47). In addition there does not appear to be a sub-change room included in the design which will restrict the transfer of contamination out of the area.

398 As a result of concerns about the potential levels of surface and airborne contamination in the Fuel Handling Area, I have raised an Assessment Finding (**AF-AP1000-RP-09**) in order to require a future licensee to justify the exposure of personnel working in the area to contamination, and to substantiate that this exposure has been reduced so far as is reasonably practicable:

> ***AF-AP1000-RP-09:*** *The licensee shall provide a justification of the exposure of workers in the Fuel Handling Area to radioactive surface and airborne contamination, including substantiation that this exposure has been reduced so far as is reasonably practicable. This finding shall be addressed before first fuel load.*

4.5.2 Normal Operation – Contaminated Areas - Conclusions

Westinghouse has demonstrated that it has incorporated engineering controls into the AP1000 design which will minimise the generation of contamination and also the potential for it spreading. It has also identified measures further down the hierarchy of contamination control measures which can be utilised by future licensees. However, Westinghouse has not provided sufficient information in order to provide assurance that 'defence in depth' for contamination control has been applied throughout the AP1000 design. A human factors assessment of Westinghouse's arrangements should be used to substantiate its effectiveness. Administrative controls will contribute to any contamination control arrangements, and it is acknowledged that these be determined by the licensee at the site specific phase.

4.5.3 Normal Operation – Contaminated Areas - Findings

399 The current submission from Westinghouse does not demonstrate 'defence in depth' with regard to the control of contamination, and so the following Assessment Finding has been raised:

> ***AF-AP1000-RP-08:*** *The licensee shall provide a justification that substantiates that defence in depth has been applied to the contamination control aspects of the AP1000 design, taking into account the response to AF-AP1000-RP-07. In particular, the report*

shall demonstrate that there are suitable and sufficient barriers to the spread of contamination from controlled areas to non-designated areas. The justification shall be supported by a suitable and sufficient human factors assessment. This finding shall be addressed before fuel on-site.

400 There are concerns about the potential levels of surface and airborne activity in the Fuel Handling Area, and so the following Assessment Finding has been raised:

AF-AP1000-RP-09: *The licensee shall provide a justification of the exposure of workers in the Fuel Handling Area to radioactive surface and airborne contamination, including substantiation that this exposure has been reduced so far as is reasonably practicable. This finding shall be addressed before first fuel load.*

4.6 Normal Operation - Ventilation

401 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of ventilation would include the following matters.

- Airborne contamination.
- Ventilation to allow access into the Containment Building at power.
- Radiation exposures incurred during maintenance and testing.
- Control of naturally-occurring radon.

4.6.1 Normal Operation – Ventilation - Assessment

402 The assessment of ventilation systems within the AP1000 was undertaken by ND mechanical engineering assessors, and the scope and findings of that assessment are described in the Step 4 Mechanical Engineering Assessment of the AP1000 Division 6 Assessment Report (Ref. 74). I liaised with the assessors in order to ensure that radiological protection aspects were considered in their assessment.

403 Future licensees will be expected to demonstrate that they have reduced the exposure of personnel to radon gas so far as is reasonably practicable. Practical measures to control naturally-occurring radon gas are dependent on the geological characteristics of the site, and so were not considered in my report. However, exposure to radon should be considered in radiological risk assessments at the site specific phase.

4.6.2 Normal Operation – Ventilation - Conclusions

404 The mechanical engineering assessors are responsible for assessing this topic area. However, I have no concerns from the evidence presented.

4.6.3 Normal Operation – Ventilation - Findings

405 There are no specific assessment findings for this topic.

4.7 Normal Operation – Radiological Instrumentation

406 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of the RMS would include the following matters.

- Control and instrumentation for monitoring direct radiation and contamination throughout the plant, including ponds.
- Radiation exposures incurred during maintenance and testing.

4.7.1 Normal Operation – Radiological Instrumentation - Assessment

407 Early in Step 4, Westinghouse presented information on its Radiation Monitoring System (RMS), which provided confidence that it had incorporated robust arrangements for monitoring radiological conditions in the generic AP1000 design. When considering this information, and the fact that the selection of instrumentation for radiological monitoring will be the responsibility of future licensees, I decided not to focus significant assessment effort in this area. However, Westinghouse has provided useful information on the RMS in the PCSR (Ref. 12), and a brief summary is discussed hereafter.

408 Westinghouse claims that the RMS provides airborne monitoring, plant effluent monitoring, process fluid monitoring, and continuous indication of the radiation environment in plant areas where such information is needed.

409 Permanent area gamma monitors are positioned in a number of areas, including the Primary Sampling Room, Operating Deck within containment, the Main Control Room, the Fuel Handling Area, the Radwaste Building and the Hot Machine Shop. With the exception of the primary sampling room, all monitors have a nominal range of 1 μSvh^{-1} to 100 $mSvh^{-1}$. The primary sampling room monitor has an extended upper range to 100 Svh^{-1} for use as a post-accident monitor. The monitors in the Fuel Handling Area can detect radiation from a fuel criticality accident in the areas occupied by personnel where fuel is stored and handled, with indications and alarms in the Main Control Room.

410 A number of airborne and fluid activity monitors are listed for detecting and monitoring incidents such as leaks and other releases. Additional area monitors are provided for post-accident monitoring. The majority of these systems initiate alarms or display data in the Main Control Room so that information is obtained remote to the affected areas.

411 The use of portable (i.e. hand-held) radiological instruments is an operational issue and is out of scope. The specification of fixed radiological instrumentation (e.g. manufacturer) used as part of the RMS is also out of scope.

4.7.2 Normal Operation – Radiological Instrumentation - Conclusions

412 From the information provided, I have no concerns regarding the RMS. Westinghouse has designed the system to monitor radiological conditions and to identify incidents with the potential to lead to elevated doses both on and off-site. My assessment has not examined the operator doses involved with maintaining and calibrating installed equipment because these factors will depend on the exact specification of the instruments, and this will be determined at the site specific phase.

4.7.3 Normal Operation – Radiological Instrumentation - Findings

413 There are no assessment findings for this topic.

4.8 Normal Operation - Decontamination

414 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of decontamination would include the following matters.

- Facilities for decontamination of employees, articles (e.g. easily portable items, large items, personal protective equipment) and areas of the facility.
- Decontamination of shielded enclosures and manipulation systems.
- Facilities for decontamination during accidents and decontamination during accidents and decommissioning.

4.8.1 Normal Operation - Decontamination - Assessment

415 During Step 4, ND and the Environment Agency were considering Westinghouse's plans for decommissioning an AP1000 reactor site, including the practicability of decontamination. Since there was a clear link with decontamination during operations, ND and the Environment Agency raised a joint RO on decontamination, (Ref. 9) that covered both operations and decommissioning. ND and the Environment Agency also raised an RO on decommissioning (Ref. 9).

416 In preparing the RO, ND and the Environment Agency again reviewed the information on decontamination that was contained within Westinghouse's submission and identified a number of aspects of decontamination that the GDA submission needed to address. The submission did contain some outline information on decontamination, but the following additional information was requested in order to enable a robust assessment of Westinghouse's arrangements:

- Detail on the baseline decontamination strategy for an AP1000 during operations and maintenance, including;
  - Detail on the predicted decontamination requirements during operations and maintenance.
  - A baseline decontamination strategy / philosophy.
  - Detail on any design features to support decontamination.
  - Detail on any decontamination systems which are included in the GDA design.
  - Clarity on the decontamination systems and techniques which could be used by the operator, in those areas where decontamination systems are not already included within the GDA design.
  - Clarity on what level of automation will be involved.
- Detail on the baseline decontamination strategy for an AP1000 during Post Operational Clean Out (POCO) and decommissioning, including;
  - Detail on the predicted decontamination requirements during POCO and decommissioning.
  - A baseline decontamination strategy / philosophy.
  - Detail on the baseline decontamination systems and techniques assumed in the GDA design, including details on any enabling design features provided.

- Detail on the laundry provision.
- Demonstration that the wastes arising from decontamination operations have been considered, including;
  - Detail on how decontamination waste arisings have been minimised through both the design and the operational, maintenance and decommissioning philosophies developed for an AP1000.
  - Consistency with the environmental submissions, e.g. the discharge assessment and the disposability assessment.

417 Westinghouse's responses to the RO on decontamination and the RO on decommissioning (Ref. 9) were assessed by ND's TSC, REACT Engineering Limited, and the TSC's findings are reported in Ref. 43. The assessment of Westinghouse's response to the RO on decommissioning is discussed in the Step 4 assessment report on radioactive waste and decommissioning (Ref. 44).

418 The RO on decontamination had four actions associated with it on the topics listed below.

- ROA-1: Decontamination during operations and maintenance.
- ROA-2: Decontamination during POCO and decommissioning.
- ROA-3: Laundry facilities.
- ROA-4: Decontamination wastes.

419 ROA-1 (Ref. 9) requested Westinghouse to provide its baseline decontamination strategy for an AP1000 reactor during operations and maintenance. The use of decontamination techniques will have a significant impact on other aspects of the plants operation. These will include the methodologies adopted (manual or remote), the shielding and containment requirements, and the eventual decommissioning techniques used. It will also have a significant effect on the operational and decommissioning waste routes (including disposal of decontamination wastes), which is considered further under ROA 4. ND and the Environment Agency would therefore expect details on the overall decontamination strategy and techniques during operations and maintenance, which should include techniques for both individual items and complete systems. The ROA requested that Westinghouse should identify:

- Detail on the predicted decontamination requirements during operations and maintenance. This should consider any key plant areas, systems, components, etc which will regularly require decontamination during operations and maintenance (e.g. the ponds/pits).
- A baseline decontamination strategy / philosophy that could be adopted, including whether decontamination will be done in-situ or at a designated location(s).
- Detail on any design features to support decontamination, e.g. minimisation of material hold-up, material selection, space / layout, and connection points.
- Clarity on what level of automation will be involved, e.g. details on any automated / remote cleaning of the reactor pool.
- Detail on any decontamination systems which are included in the GDA design, for deployment during operations and maintenance.

- Clarity on the decontamination systems and techniques which could be used by the operator, in those areas where decontamination systems are not already included within the GDA design.

420 When addressing the latter two matters, Westinghouse was asked to consider the primary circuit, large items, size reduced and small items, plant areas, shielded enclosures, ponds/pits, manipulation systems, and decontamination in the event of accidents or abnormal operations (including decontamination of personnel).

421 ROA-2 (Ref. 9) requested Westinghouse to provide its baseline decontamination strategy for an AP1000 reactor during POCO and decommissioning. As the use of decontamination techniques may have a significant impact on other aspects of plant operation (as discussed above), ND and the Environment Agency expected to see details on the overall decontamination strategy and techniques during POCO and decommissioning, which should include techniques for both individual items and complete systems. ROA-2 (Ref. 9) requested Westinghouse's response to take account of the following matters.

- Detail on the predicted decontamination requirements during POCO and decommissioning. This should consider any key plant areas, systems, components, etc which will require POCO or decontamination as part of decommissioning.
- A baseline decontamination and POCO strategy / philosophy that could be adopted, including whether decontamination will be done in-situ or at a designated location(s).
- Detail on the baseline decontamination systems and techniques assumed in the GDA design, including details on any enabling design features provided, e.g. minimisation of material hold-up, material selection, space / layout, and connection points. This should consider the primary circuit, large items, size reduced and small items, plant areas, shielded enclosures, ponds/pits, manipulation systems, and decontamination in the event of accidents or abnormal operations (including decontamination of personnel).

422 ROA-3 (Ref. 9) requested Westinghouse to provide information regarding laundering of contaminated clothing since this was an inherent part of operating a reactor site, and was also required during both the commissioning and decommissioning stages of the lifecycle. Laundering required both contamination control and waste disposal arrangements.

423 ROA-4 (Ref. 9) requested Westinghouse to provide information regarding decontamination wastes. The assessment of Westinghouse's response was undertaken by the Environment Agency (Ref. 75).

424 The response to the RO (Ref. 9) outlined systems which were expected to become contaminated during normal and shutdown operations, including the probable frequency of decontamination. It also provides details of design features which facilitate decontamination, such as the claim that many of the systems have a surface finish which minimises adsorption of contamination and aids decontamination.

425 It appears that Westinghouse has expended efforts on determining decontamination techniques for pumps, with details on potential approaches for a range of pumps. Certain types of pumps can be decontaminated in-situ, such as the CVS Makeup Pumps, whilst others are designed for dismantling and removal for ex-situ decontamination. Portable Decontamination Units can be used for in-situ processing of materials, with temporary shielding and covers being deployed to minimise the exposure of personnel.

426 The majority of my assessment has focussed on facilities for decontamination, and Westinghouse claims that the Hot Machine Shop will be used for decontaminating small moveable items, using either the Decontamination Glove Box or Decontamination Basin. The Glove Box is capable of decontaminating components smaller than 0.6m long, and the basin is capable of handling equipment up to 2.4m x 2.4m (Ref. 9).

427 Westinghouse expects that the decontamination of large moveable equipment will normally take place in the Cask Washdown Pit. An example of this use is the decontamination of the spent fuel shipping casks which can be moved into the Cask Washdown Pit using the crane located near the Cask Loading Pit. Any other large equipment that can be transported to the rail car bay in the Auxiliary Building and is within the weight limit of the cask loading area crane can also be transferred to the Cask Washdown Pit for decontamination. Westinghouse adds that if the Cask Washdown Pit cannot be used, a temporary structure can be erected in a miscellaneous staging for the decontamination of large equipment.

428 Westinghouse outlines plans for decontamination facilities for a clean staging, decontamination, and checkout area near containment which should be used for applying and removing protective materials that are used for items such as portable tools, equipment, and instrumentation inside containment during inspection and maintenance work. The area shall also provide sufficient space to allow for segregation of dry active waste.

429 A second staging area will also be provided which shall provide sufficient space to erect special enclosures for decontamination of large equipment items which cannot be decontaminated in the Cask Washdown Pit.

430 Westinghouse provides details of additional facilities for decontaminating Low Level Waste (LLW). An enclosed booth, vented through the heating, ventilation and air conditioning (HVAC) system will be available in the Radwaste Building. It will be operated though a viewing window complete with glove and sliding ports. Waste will enter either of the units from a shared conveyor/roller table, onto a working platform. Within the units will be simple tools to perform the decontamination tasks required (e.g. dismantling, cutting, swabbing). The decontamination enclosure may also include a small wet area for washing/wiping/scrubbing with decontamination solutions.

431 Two personnel decontamination rooms are available in the Annex Building to allow for full body decontamination of workers. Two shower stalls and three sinks are provided in each room, with potentially contaminated liquor draining to the Radioactive Waste Drain System (WRS).

432 With regard to laundry facilities, Westinghouse stated that there is no laundry included in the design of the AP1000. The LLW and Very Low Level Waste (VLLW) laundry items will be bagged and collected from the change rooms. The bagged laundry will be temporarily stored within the Radwaste Building before shipment to an off-site laundry. Westinghouse identifies one facility which is available in the UK and adds that it is anticipated that the licensee will use this or an equivalent service during the plant life. It claims that it would be possible to have an on-site laundry facility should a future licensee so wish. This response is adequate to demonstrate that Westinghouse has identified a reasonable approach to the management of contaminated clothing.

433 The decontamination facilities described above demonstrate that Westinghouse has attempted to apply a philosophy of contamination control to the design. However, closer examination of the layout of areas such as the Hot Machine Shop suggests that some areas are compact with limited space available for conducting potentially hazardous

decontamination practices. This could lead to workers receiving elevated radiation exposures and increase the potential for radiological accidents occurring. My concerns in this area were discussed in Section 4.4.1.3.4, and the improvement of the design of the Hot Machine Shop is captured in Assessment Finding **AF-AP1000-RP-07**.

434 The RO requested information on the use of automation for decontamination processes, and a small amount of detail is included in the response. It states that there are no plans for automated / remote cleaning of the reactor pool in the GDA design, but that automated / remote cleaning of the reactor pool is possible if the utilities decide it is necessary. Westinghouse claims that robotic devices, such as refuelling cavity decontamination units, are considered in the layout of the refuelling cavity so that interferences such as light fixtures, tool hangers and personnel ladders are removable or do not affect the use of the robotic units (Ref. 9).

435 The TSC report on decommissioning and decontamination (Ref. 43) outlined the following conclusions on Westinghouse's approach to decontamination:

- Westinghouse has made an adequate response to this line of enquiry for the purposes of GDA, but further work may be required at the site licensing stage to help underpin the plans developed by the Licensee.
- The information on decontamination during POCO/decommissioning, provides a great deal of detail on generic techniques but no additional AP1000-specific information. The TSC infers that in the future additional engineering might be required to achieve a full POCO, unless additional detail can be provided to the contrary.
- There is some lack of clarity over which items have decontamination capability provided for and which may not, but this has not caused excessive concern because confidence has been gained through those examples where detail has been provided.
- There is a lack of clarity with regard to the techniques used for in-situ decontamination of some items, but the TSC acknowledges that this is a detailed issue which does not necessarily need to be resolved during GDA.
- There are some concerns over the size of facilities allowed for in the design, such as the hot workshop.

4.8.2 Normal Operation - Decontamination - Conclusions

436 Based on the evidence assessed, I am satisfied that Westinghouse has adequately demonstrated that it has taken account of decontamination practices, and applied measures to restrict the exposure of personnel involved with decontamination activities. My principal concern regarding the inadequate size of the Hot Machine Shop is captured in Assessment Finding **AF-AP1000-RP-07**.

4.8.3 Normal Operation - Decontamination - Findings

437 No specific assessment findings were identified for this topic.

4.9 Normal Operation – Waste Handling and Decommissioning

438 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of waste handling and decommissioning would include the following matters.

- Control of direct radiation and contamination, and application of ALARP.
- Management of doses during waste handling and storage.

439 The assessment of radioactive waste and decommissioning was undertaken by ND's radioactive waste and decommissioning assessors, and the scope and findings of that assessment are described in the Step 4 Radioactive Waste and Decommissioning Assessment of the AP1000 Division 6 Assessment Report (Ref. 44). My assessment was targeted to the radiological protection aspects of handling radioactive waste, especially since handling radioactive waste is also a key radiological protection factor during decommissioning. I did not assess decommissioning other than with regard to handling radioactive waste.

4.9.1 Normal Operation – Waste Handling and Decommissioning – Assessment

440 My assessment of worker doses for waste handling is detailed in Section 4.4.1.3.3.3, including Assessment Finding **AF-AP1000-RP-05,** regarding a requirement to revisit the dose estimate for this activity.

441 The main focus of my assessment for this topic has been the adequacy of waste processing facilities with regard to minimising the radiological risk to waste handling personnel. My assessment has particularly focussed on the equipment and layout within the Radwaste Building, and my concerns in this area have been outlined in Section 4.4.1.3.4.

442 In particular, the original layout of the Radwaste Building provided by Westinghouse in its response to a TQ contained many of the features associated with an effective radioactive waste management programme, but restrictions on space meant that potentially hazardous practices would not be segregated. I was concerned that it would be difficult to protect workers who are undertaking one task from being exposed to radiation from others, and in particular, that the layout did not facilitate effective contamination control.

443 Westinghouse has submitted a new layout as part of its response to the RO on health physics and radioactive waste facilities which demonstrates improvements to the design, such as better segregation of activities and incorporation of contamination control features such as a sub-change room (Ref. 9). However, there are still concerns regarding the amount of space available, and these are the subject of the radioactive waste and decommissioning assessment (Ref. 44).

4.9.2 Normal Operation – Waste Handling and Decommissioning - Conclusions

444 Westinghouse has revisited the layout of the Radwaste Building and demonstrated an understanding of ND's concerns regarding segregation of hazardous activities and contamination control. However, further development work will be required by the licensee to demonstrate that the layout demonstrated ALARP and that there is adequate space for conducting waste processing tasks safely. As such, the radioactive waste and decommissioning assessors have raised an assessment finding (Ref. 44)

4.9.3 Normal Operation – Waste Handling and Decommissioning - Findings

445 The assessment finding for this topic area is captured in the radioactive waste and decommissioning assessment report (Ref. 44). Assessment Finding **AF-AP1000-RP-05** regarding doses from waste processing is described in Section 4.4.1.3.3.3.

4.10 Normal Operation - Public Exposure

446 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of public exposure would include the following matters.

- Liaison with the Environment Agency on optimisation of doses to the public from direct radiation originating within the site boundary (ND has the lead).
- Liaison with the Environment Agency on optimisation of doses to the public from authorised discharges (the Environment Agency has the lead).

447 As explained in Section 2.2 above, the regulation of public radiation exposure during normal operation is shared between the Environment Agency and HSE, where IRR99 (Ref. 19) is enforced by ND on behalf of HSE, and EPR10 (Ref. 30) is enforced by the Environment Agency. IRR99 (Ref. 19) require dose constraints to restrict exposure to ionising radiation at the planning stage where it is appropriate to do so. The guidance to IRR99 (Ref. 22) 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. 4) in relation to NT.1 Target 3, and advises that HSE's view is that a single source should be interpreted as a site under a single dutyholder's control, since this is an entity for which radiological protection can optimised as a whole. However, the former Health Protection Agency's Centre for Radiation, Chemical and Environmental Hazards (HPA-CRCE) has recently recommended that the dose constraint for members of the public from new NPPs should be 0.15 mSv per year (Ref. 24).

448 Section 4.4.1.1.2 discusses Westinghouse's performance against Target 3, and from Westinghouse's response it is clear that doses from direct radiation will be below the BSO of 20 μSvy^{-1}. This is accomplished as a result of the deployment of effective shielding around the reactor and its containment, as outlined in Section 4.3.1.6.1.

449 Throughout Step 4, radiological protection assessors and radioactive waste and decommissioning assessors from ND have jointly attended meetings with assessors from the Environment Agency on topics which have common interest, such as radioactive waste, decommissioning and decontamination. As such, I have liaised with the Environment Agency on matters regarding public doses which are outlined in its assessment report (Ref. 75).

4.10.1 Normal Operation - Public Exposure - Conclusions

450 Westinghouse has demonstrated that public exposure from direct radiation under normal conditions has been reduced so far as is reasonably practicable.

4.10.2 Normal Operation - Public Exposure - Findings

451 There are no assessment findings associated with this topic area.

4.11 Accident Conditions – Persons On-site

4.11.1 Accident Conditions – Persons On-site - Assessment

452 Table 3 of my Step 4 Plan (Ref. 1) explained that my assessment of impacts to people on-site during accidents, including criticality control, would include the following matters.

- Radioactive source term, dose rates and shielding.
- Consideration of areas of the facility, radiation levels, airborne contamination levels, and exposure / evacuation times.
- Criticality accidents, including criticality control in fuel ponds and in dry storage of spent fuel (see Appendix 1 of this report).
- Facilities and design features for responding to accidents.
- Optimisation of doses to people on the site.

Impacts to people off-site during accidents are covered by Level 3 PSA, which is reported in the Step 4 Probabilistic Safety Analysis of the Westinghouse AP1000 assessment report (Ref. 31).

4.11.1.1 Criticality Accidents

453 A significant amount of time was dedicated to the assessment of criticality accidents, and particularly criticality control of the SFP. The scope, expectations and findings of the assessment are reported in Appendix A to this report.

454 The results of the assessment were that certain aspects of Westinghouse's criticality safety case have not met ND's expectations, and these matters were considered to be sufficiently serious to require the raising of GDA Issue **GI-AP1000-RP-01**. This is identified in Appendix A and Annex 2. The complete GDA Issue and associated action are formally defined in Annex 2.

> ***GI-AP1000-RP-01**. Westinghouse has not adequately demonstrated why it is not reasonably practicable to design the AP1000 SFP such that criticality control is achieved through geometrical control and fixed poisons alone.*

455 In addition, a number of assessment findings pertaining to criticality accidents have been raised, and these are described in Appendix A and Annex 1.

4.11.1.2 Non-criticality Accidents

456 A limited assessment of non-criticality assessments has been conducted. Aspects of the submission which were provided in order to support the Fault Studies assessment have been reviewed in order to assess relevant targets in the SAPs (Ref. 4). This information is principally contained in Chapter 9 of the PCSR (Ref. 12).

4.11.1.2.1 Design Basis Fault Sequences – NT.1 Target 4

457 NT.1 Target 4 in the SAPs (Ref. 4) provides frequency targets for a range of effective doses received by any person arising from a design basis fault sequence, and applies to people on the site and to members of the public off the site.

Table 8: SAPs NT.1 Target 4

Design basis fault sequences – any person	**Target 4**
The targets for the effective dose received by any person arising from a design basis fault sequence are: **On-site** BSL: 20 mSv for initiating fault frequencies exceeding 1 x 10^{-3} pa 200 mSv for initiating fault frequencies between 1 x 10^{-3} and 1 x 10^{-4} pa 500 mSv for initiating fault frequencies less than 1 x 10^{-4} pa BSO: 0.1 mSv **Off-site** BSL: 1 mSv for initiating fault frequencies exceeding 1 x 10^{-3} pa 10 mSv for initiating fault frequencies between 1 x 10^{-3} and 1 x 10^{-4} pa 100 mSv for initiating fault frequencies less than 1 x 10^{-4} pa. BSO: 0.01 mSv	

458 Guidance on NT.1 Target 4 on design basis fault sequences for any person on or off the site is in Paras 598 to 601 of the SAPs. Guidance on radiological analysis of fault conditions is provided in TAG T/AST/045 (Ref. 18) in Paras 4.1 to 4.8, and 4.17 to 4.19. Guidance on radiation protection during accident conditions is provided in RP.2 and Paras 480 – 483 of the SAPs.

459 Radiological consequences of design basis events to members of the public off the site are considered in the assessment report regarding Step 4 Fault Studies – Design Basis Faults Assessment of the AP1000 (Ref. 76).

460 Site specific calculations for design basis radiological consequences are out of scope of GDA. However to gain confidence that acceptable AP1000 site specific calculations will be possible in the future, it is necessary to know for GDA that radiological consequences predicted for a generic site can be compared favourably with the established UK limits. In order to obtain generic information, the fault studies assessors raised an RO (Ref: 9) which required it to recalculate the radiological consequences for design basis faults using methods and assumptions consistent with relevant UK good practice and to explicitly compare the results against the appropriate Target 4 limits.

461 Westinghouse's response (Ref. 9) analysed the radiological consequences of the following faults. The calculated dose received by the most exposed worker on-site is also provided and compared to the relevant BSL from Target 4. The doses for a Main Steam Line Break and a Feedline Break are taken from Chapter 9 of the PCSR (Ref. 12), and reproduced below.

Table 9: On-site Doses from Specified AP1000 Faults

Fault	Worker Dose (mSv)	Relevant BSL (mSv)
SBLOCA	6	200
LBLOCA	15	100
Rod Ejection Accident	8	500
Locked RCP Rotor	11	500
Main Steam Line Break	78	500
SGTR	5	20
Small Line Break Outside of Containment	2	20
Loss of Off-site Power	2	20
Single Rod Withdrawal	11	500
Feed-line Break	78	200

462 The calculated worker doses meet the BSL, but are all above the BSO of 0.1 mSv. As a result, the risk associated with these events may not be ALARP, and a formal ALARP justification would be expected.

463 The Step 4 Fault Studies – Design Basis Faults Assessment of the AP1000 (Ref. 76) concluded that it should be possible for future site specific analysis of design basis faults to show compliance with Target 4 of the SAPs. The fault studies assessors did note that Westinghouse needs to provide further discussion on the radiological consequences of shutdown faults (Ref. 76).

464 Chapter 9 of the PCSR (Ref. 12) also contains limited information on non-reactor faults (i.e. faults that are unrelated to the reactor but that may still occur within the facility). The following faults were identified but screened out from further analysis:

- Over-raising fuel in refuelling cavity or SFP.
- Exposure in areas contaminated by pool water.
- Unauthorised operator entry into active areas/areas of high dose rate.
- Inappropriate handling or use of other sources.
- Failure to adequately control chemistry to manage dose rates.
- Operator exposure from stored waste because of inadequate shielding or waste consignment error.
- Operator exposure from incorrect consignment of waste because of assessment, clerical, or sampling errors.

465 While some of the faults can not be adequately assessed during GDA, such as the inappropriate use of radiography sources, which is a site specific issue, it is my opinion that the arguments for not developing the study of some of these faults further is not robust, and some fault scenarios could potentially have been partially addressed at the

generic design stage by the inclusion of engineering controls. Nevertheless, these faults can be reviewed again at the site specific stage.

466 One fault which I consider to have warranted further attention in Westinghouse's submission during GDA is the access of areas of high dose rate by unauthorised personnel. Chapter 9 of the PCSR (Ref. 12) states that access to very high radiation areas will be controlled by administrative means by the licensee, whereas there could potentially be scope to use physical barriers to prohibit access to unauthorised personnel. The information in Chapter 12 of the EDCD conflicts with that in the PCSR by stating that high and very high radiation areas are either locked or barricaded (Ref. 60). I have raised an Assessment Finding **AF-AP1000-RP-10** to address concerns in this matter.

> ***AF-AP1000-RP-10**: The licensee shall provide a report that demonstrates that the hierarchy of control measures has been appropriately applied with regard to restricting access to high radiation areas to authorised personnel only. This finding shall be addressed before fuel on-site.*

467 The non-reactor faults that were progressed for further analysis by Westinghouse involve an operator falling into the flooded Refuelling Cavity or SFP, and a fall in water level in the Refuelling Cavity or SFP, with an associated loss of shielding.

468 For faults involving an operator falling into the SFP or the flooded Refuelling Cavity, Westinghouse calculated the maximum total dose to the operator of 87 µSv, based on a bounding case of an operator falling into the SFP during spent fuel movements when the fuel is in closer proximity to the pool surface and so the amount of shielding afforded by the water is at a minimum (Ref. 12). This dose is comprised of two components:

- Exposure to direct radiation, assuming a five minute exposure at the surface of the pond, with an assumed dose rate of 200 μSvh^{-1}. This component of the dose is 17 µSv.
- Dose incurred as a result of ingesting pool water. Westinghouse assumed that 5 ml of water is ingested and that it is contaminated with tritium and cobalt-60, leading to a total committed effective dose of 70 µSv.

469 Westinghouse identifies measures that will reduce the risk of an operator falling into the SFP or Refuelling Cavity, such as physical barriers (walls and railings), and the use of harnesses where appropriate. It adds that lifebuoys and equipment suitable for retrieving a person in the water, such as suitable egress ladders, shepherds' hooks, and poles, should be provided.

470 I do not consider the doses identified by Westinghouse for this fault to be of concern, and the measures identified to minimise the radiological risks to operators working around the SFP and Refuelling Cavity from this fault scenario appear reasonable. However, I have not assessed whether the data used in the calculations or the calculation methods themselves are appropriate.

471 For faults involving the loss of water in the SFP and Refuelling Cavity, Westinghouse claims that the radiological consequences for workers would be very low, as the fault would activate alarms in the MCR and workers would be excluded while the water level is topped up (Ref. 12). When reviewed in isolation, this claim appears reasonable, but it is possible that fault scenarios that could lead to a decrease in water levels could be more complex than Westinghouse has indicated, and so I cannot make any firm conclusions on this matter.

472 It is not entirely possible, nor appropriate, to carry out a detailed assessment of faults which could lead to significant doses to persons on-site before the detailed design is finalised and the exact operating regime is determined.. The radiological consequences associated with reasonably foreseeable accidents and, in particular, the adequacy of engineering controls and design features which are intended to reduce the exposure of personnel so far as is reasonably practicable, will need to be assessed in detail at the site specific phase. Consequently, I have raised Assessment Finding **AF-AP1000-RP-11** in order to capture this matter:

> ***AF-AP1000-RP-11**: The licensee shall provide a safety case that demonstrates that the on-site specific radiological consequences analyses for design basis events (including hazards) are acceptable and have taken due cognisance of usual UK methodology assumptions, and have explicitly compared the results of those analyses against NT.1 Target 4 regarding the predicted initiating fault frequency versus dose to individuals on the site in the SAPs. This finding shall be addressed before fuel on-site.*

4.11.1.2.2 Impacts on-site of Accidents – NT.1 Targets 5 and 6

473 Target 5 of the SAPs (Ref. 4) concerns the individual risk of death to a person on-site from on-site accidents that result in exposure to ionising radiation. Target 6 of the SAPs (Ref. 4) concerns the frequency of any single accident for specific on-site dose bands. These targets do not cover intervention personnel.

Table 10: SAPs NT.1 Target 5

Individual risk of death from on-site accidents – any person on the site	Target 5
The targets for the individual risk of death to a person on the site, from on-site accidents that result in exposure to ionising radiation, are: BSL: 1×10^{-4} pa BSO: 1×10^{-6} pa	

Table 11: SAPs NT.1 Target 6

Frequency dose targets for any single accident – any person on the site		Target 6
The targets for the predicted frequency of any single accident in the facility, which could give doses to a person on the site, are:		
Effective dose, mSv	**Predicted frequency per annum**	
	BSL	**BSO**
2 – 20	1×10^{-1}	1×10^{-3}
20 – 200	1×10^{-2}	1×10^{-4}
200 – 2000	1×10^{-3}	1×10^{-5}
> 2000	1×10^{-4}	1×10^{-6}

474 Westinghouse has not explicitly provided details of the AP1000 design's performance when compared to Targets 5 and 6. This topic will need to be assessed in detail at the site specific phase, and I have raised Assessment Finding **AF-AP1000-RP-12** in order to capture this matter:

> ***AF-AP1000-RP-12****: The licensee shall provide a safety case that demonstrates that the on-site specific radiological consequences analyses for accidents (including hazards) are acceptable and have taken due cognisance of usual UK methodology assumptions, and have explicitly compared the results of those analyses against NT.1 Target 5 regarding the risk impact to individuals from all the facilities on the site, and against NT.1 Target 6 regarding the predicted single accident frequency versus dose to individuals on the site in the SAPs. This finding shall be addressed before fuel on-site.*

4.11.1.3 Post-Accident Accessibility

4.11.1.3.1 Doses Incurred by Workers Involved in Post-Accident Activities

475 In Chapter 24 of the PCSR, Westinghouse states that following a severe accident, the use of passive systems for mitigation of major accidents maintains the integrity of the reactor containment for 72 hours without standby systems. However, the main control room (MCR) will continue to be manned and some operations will be undertaken during this period to monitor and prepare to maintain the passive containment water inventory and SFP cooling (Ref. 12).

476 Westinghouse states that the gamma ray source term following an accident increases to a maximum after approximately 2 hours and then decreases as a result of radioactive decay (Ref. 12). The containment sources for various times are shown in the AP1000 Radiation Analysis Design Manual (Ref. 45).

477 Calculations have been carried out on MCR occupancy and ingress/egress and four other specific operations that are expected to be needed following a worst-case accident (Ref. 12):

- MCR occupancy.
- MCR ingress and egress.
- SFP Makeup Valve Alignment.
- Temporary Water Hook-up to PCS.
- Ventilation control for temporary HVAC to MCR and Control and Instrumentation (C&I) Equipment Room.
- Provision of temporary power to transformers in Electrical Equipment Rooms.

478 The maximum dose rate in the MCR following an accident has been calculated to require classification as Zone IV, i.e., <1 $mSvh^{-1}$. The dose rate falls rapidly from this maximum as the short-lived fission products decay (Ref. 12).

479 MCR operations are assumed to run on 12-hour shifts, beginning at the time of the accident and running for 30 days thereafter. The total integrated dose for this activity is provided as 2.7 mSv (Ref. 12). A crew that accesses the MCR for a 12-hour shift each day for 30 days would receive 3.6 mSv according to Westinghouse's estimates (Ref. 12), based on a single ingress and egress operation taking approximately 5 minutes. Hence,

the total dose to a shift team in the MCR (30 ingress and egress operations at 12-hour intervals, and 30 12-hour shifts in the MCR) is 6.3 mSv. Westinghouse states that this number is conservative as it does not allow for relief shifts. Westinghouse claims that it has provided shielding and effective ventilation in order to permit access to, and prolonged occupancy in the MCR. The ventilation provision for the MCR has been assessed by the mechanical engineering assessors (Ref. 74).

480 The highest dose from a post-accident vital area access operation (SFP makeup valve alignment after 64 hours after the accident) is given as 11.1 mSv (Ref. 12). However, this falls to 5.4 mSv if delayed to 168 hours (1 week) after the accident. The remaining activities are carried out 64 hours after the accident and involve lower worker doses.

481 The doses highlighted above suggest that the measures have ensured that doses comply with the levels for intervention personnel as required by REPPIR (Ref. 37), but there is an overriding responsibility to demonstrate ALARP, and I have not assessed whether additional control measures could have been adopted to minimise potential post-accident worker doses further. I have also not made significant efforts to identify whether there are any additional tasks which are likely to be undertaken following an accident which could lead to a significant exposure. However, Westinghouse states that samples may need to be collected from the Primary Sampling Room following an accident, but no details on exposure are included in Chapter 24 of the PCSR.

4.11.1.3.2 Post-Accident Zoning

482 The table titled ‘Radiological Classification of Areas and Access Requirements’ in Chapter 24 of the PCSR (Ref. 12) describes the zoning of areas within Containment and within the Auxiliary Building during accident conditions.

483 All areas within Containment are designated as Zone IX (>5 Svh^{-1}) following an accident, meaning that access is prohibited using Westinghouse’s criteria (Ref. 12). Certain areas of the Auxiliary Building are designated with a zone for a post-accident phase, and a number of these areas are specifically highlighted as requiring access, including part of the Maintenance Floor Staging Area which has the highest zoning designation of VIII (between 1 and 5 Svh^{-1}). Westinghouse is correct to identify that access to this area would need to be extremely restricted, but the reasons for requiring access to this area are not clear. Assessment Finding **AF-AP1000-RP-13** addresses this matter:

> ***AF-AP1000-RP-13**: The licensee shall provide a safety case to identify the specific areas which are likely to require access during a post-accident phase, and to identify potential doses to workers carrying out those activities and demonstrate that they are ALARP. This finding shall be addressed before the first fuel load.*

484 Chapter 12 of the European Design Control Document (EDCD) (Ref. 60) contains a series of layout diagrams that illustrate routes for post-accident worker access, including area designations. This information appears useful for determining the exposure of workers undertaking post-accident duties, although I have not compared it to the operator dose and area zoning information presented in Chapter 24 of the PCSR. This topic will be the subject of further assessment as a result of Assessment Finding **AF-AP1000-RP-13**.

4.11.1.3.3 Emergency Facilities

485 The AP1000 design does not incorporate dedicated emergency access facilities. Westinghouse states that, following a design basis accident, entry arrangements will be determined according to conditions at the time. It adds that suitable forward control points are likely to be established at low dose rate locations. The post-accident radiation zoning for the standard, operational health physics facilities at the access point to the RCA in the Annex Building is given as Zone VI (≤100 $mSvh^{-1}$) in Ref. 12, and so prolonged access to these areas may not be possible.

486 While it is good practice at UK nuclear facilities to have dedicated facilities for responding to emergencies, which contain design features designed to address radiological protection requirements specific to large-scale radiation accidents, it is acknowledged that such designs can only be finalised at the site-specific phase. As a result, this matter has been included within the scope of Assessment Finding **AF-A1000-RP-07** which concerns the design of health physics facilities outlined in Section 4.4.1.3.4.

4.11.1.3.4 Post Accident Monitoring

487 The PCSR provides examples of post-accident area monitors (Ref. 12):

- Containment high-range radiation monitors – These four monitors measure the radiation from the radioactive gases in the containment atmosphere. The data are displayed in the MCR. Alarms are provided in the MCR and signals to the protection and safety monitoring system for containment air filtration isolation and RNS valve closure and containment isolation. The monitors have a nominal range of 10 $mSvh^{-1}$ to 105 Svh^{-1}.
- Primary Sampling Room area monitor – The primary sampling station is the location where samples are collected after a postulated accident. The monitor provides local readout, and audible and visual alarms are visible upon entry into the sampling room. Indication and alarm are also provided in the MCR.
- Control Support Area (CSA) monitor – Located in the Annex Building, outside the RCA, the CSA is the location from which engineering support will be provided to the operators following a postulated accident. A local readout, and audible and visual alarms are visible upon entry into the CSA. Indication and alarm are also provided in the MCR.

488 While the exact type and location of radiological instrumentation will be selected by a future licensee at the site specific phase, Westinghouse has provided sufficient information to provide confidence that it has identified suitable locations for monitoring radiological conditions in order to assess the level of risk to on-site personnel involved with post-accident tasks.

4.11.2 Accident Conditions – Persons On-site - Conclusions

489 From the evidence presented, I judge that Westinghouse has demonstrated compliance with the BSLs outlined in Target 4 of the SAPs, although a licensee will be required to demonstrate that exposures are ALARP at the site specific phase. I expect that the radiological consequences for persons on-site resulting from additional faults which have not been considered here will also need to be determined.

490 Westinghouse has demonstrated that it has made an assessment of doses which are likely to be received by workers involved in post-accident activities, and has considered access routes and monitoring requirements in order to reduce the exposure of these personnel.

491 I have not assessed the effectiveness of design features intended to minimise doses received by workers involved in post-accident activities. However, detailed assessment has demonstrated that Westinghouse's shielding arrangements have been adequately designed for normal conditions, and so I have no reason to suspect that the bulk shielding arrangements for fault and post-accident conditions are deficient.

4.11.3 Accident Conditions – Persons On-site - Findings

492 Westinghouse states that access to high radiation areas will be controlled by administrative means by the licensee, whereas there could potentially be scope to use physical barriers to prohibit access to unauthorised personnel:

> ***AF-AP1000-RP-10****: The licensee shall provide a report that demonstrates that the hierarchy of control measures has been appropriately applied with regard to restricting access to high radiation areas to authorised personnel only. This finding shall be addressed before fuel on-site.*

493 The range of faults considered during radiological consequence analysis for persons on-site will need to be extended at the site specific phase, and the licensee will be expected to demonstrate that doses to persons on-site are ALARP:

> ***AF-AP1000-RP-11****: The licensee shall provide a safety case that demonstrates that the on-site specific radiological consequences analyses for design basis events (including hazards) are acceptable and have taken due cognisance of usual UK methodology assumptions, and have explicitly compared the results of those analyses against NT.1 Target 4 regarding the predicted initiating fault frequency versus dose to individuals on the site in the SAPs.*

494 Westinghouse has not explicitly provided details of the AP1000 design's performance when compared to Target 6. This topic will need to be assessed in detail at the site specific phase, and I have raised Assessment Finding **AF-AP1000-RP-12** in order to capture this matter.

> ***AF-AP1000-RP-12****: The licensee shall provide a safety case that demonstrates that the on-site specific radiological consequences analyses for accidents (including hazards) are acceptable and have taken due cognisance of usual UK methodology assumptions, and have explicitly compared the results of those analyses against NT.1 Target 5 regarding the risk impact to individuals from all the facilities on the site, and against NT.1 Target 6 regarding the predicted single accident frequency versus dose to individuals on the site in the SAPs. This finding shall be addressed before fuel on-site.*

495 It is not clear whether the doses incurred for all post-accident work activities have been adequately estimated:

> ***AF-AP1000-RP-13****: The licensee shall provide a safety case to identify the specific areas which are likely to require access during a post-accident phase, and to identify*

potential doses to workers carrying out those activities and demonstrate that they are ALARP. This finding shall be addressed before fuel on-site.

4.12 Overseas Regulatory Interface

496 HSE's Strategy for working with overseas regulators is set out in (Ref. 77) and (Ref. 78). In accordance with this strategy, HSE collaborates with overseas regulators, both bilaterally and multi-nationally.

4.12.1 Bilateral Collaboration

497 HSE's Nuclear Directorate (ND) has formal information exchange arrangements to facilitate greater international co-operation with the nuclear safety regulators in a number of key countries with civil nuclear power programmes. These include the following:

- the US Nuclear Regulatory Commission (NRC)
- the French L'Autorité de sûreté nucléaire (ASN)
- the Finnish STUK

4.12.2 Multilateral Collaboration

498 ND collaborates through the work of the IAEA and the OECD NEA. ND also represents the UK in the Multinational Design Evaluation Programme (MDEP) - a multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities tasked with the review of new reactor power plant designs. This helps to promote consistent nuclear safety assessment standards among different countries.

499 In my radiological protection assessment, information was shared with the following overseas regulators through a series of interface meetings as follows.

- The US NRC, where we shared information on radiological protection during operation of PWRs and criticality control in spent fuel ponds.
- The Swedish Nuclear Safety Authority (SSM), where we shared information on ND's GDA process, radiological protection during operation of PWRs and criticality control in spent fuel ponds.

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

4.13 Interface with Other Regulators

501 I have worked closely with the Environment Agency through the whole of GDA. Future operators of the AP1000 will require a permit from the Environment Agency to make discharges of radioactivity into the environment and dispose of radioactive wastes. Working closely with the Environment Agency has been important since doses to members of the public during normal operation arise from discharges (regulated by the

Environment Agency) and direct radiation originating within the site boundary (regulated by ND). Also, within the workplace, there are close interfaces between radiological protection and radioactive wastes regarding topics such as decontamination, decommissioning and waste handling.

502 Working closely with the Environment Agency meant raising joint ROs, holding joint meetings with Westinghouse, undertaking a number of benchmarking visits and reviewing our respective assessments. I have ensured that ND's TSCs on radiological protection, NT and TÜV SÜD, were aware of the Environment Agency's roles and responsibilities when undertaking their work.

4.14 Other Health and Safety Legislation

503 In addition to the legislation identified in Section 2.2, a number of other pieces of health and safety legislation are also relevant to radiological protection. One such key piece of legislation is the Management of Health and Safety at Work Regulations 1999 (MHSWR99) (Ref. 29) as amended, and its ACOP and guidance (Ref. 79). This piece of legislation is particularly important since the requirement for a prior risk assessment for undertaking a new work activity involving the use of radioactive substances is in IRR99 (Ref. 19), whereas the general requirement to review and revise all occupational risk assessments is in MHSWR99 (Ref. 29).

5 CONCLUSIONS

504 This report presents the findings of the Step 4 Radiological Protection assessment of the Westinghouse AP1000 reactor.

505 To conclude, I am broadly satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for the Radiological Protection. I consider that from a Radiological Protection view point, the Westinghouse AP1000 design is suitable for construction in the UK. However, this conclusion is subject to satisfactory progression and resolution of the GDA Issue that I have identified and which is to be addressed during the forward programme for this reactor, and assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

5.1 Key Findings from the Step 4 Assessment

506 A key aspect of my assessment involved determining whether Westinghouse had applied the hierarchy of control measures, with particular emphasis on reducing radiation source terms and utilising engineering controls to restrict the exposure of workers to ionising radiation so far as is reasonably practicable. These matters are a principal concern for GDA, since administrative controls and other measures which are further down the hierarchy can only be finalised at the site specific phase by a duty holder.

507 In my opinion, from the evidence provided, Westinghouse has designed the plant to ensure that engineering controls have been incorporated into the plant in order to restrict exposures of workers to ionising radiation so far as is reasonably practicable during normal operation. It has reduced source terms and simplified its design in order to reduce the number of components within containment, such as pumps and valves, which would require inspection and maintenance, and so has reduced operator doses resulting from these tasks. Importantly, it has also demonstrated that the bulk shielding design has been systematically optimised and is adequate for both reducing the exposure of workers and also ensuring that doses incurred by members of the public resulting from direct radiation will be negligible.

508 Westinghouse has assessed the radiological protection performance of existing PWRs, and incorporated design features which represent relevant good practice in reducing doses further, such as utilising design features which reduce the time taken to perform refuelling tasks. Westinghouse has also demonstrated that the design accommodates the use of robotic technologies and remote techniques, such as those described for SG and IHP inspection tasks.

509 There are certain aspects of the AP1000 submission which did not meet my expectations, with the most serious involving the control of criticality in the SFP which is the subject of a GDA Issue. However, it is my opinion that the matters of concern which have been captured in Assessment Findings, such as the inadequate design of health physics facilities, should not be excessively complex to resolve, and Westinghouse has been proactive in suggesting potential strategies to resolve some of them.

5.1.1 Assessment Findings

510 I conclude that the Assessment Findings listed in Annex 1 should be programmed during the forward programme of this reactor as normal regulatory business.

5.1.2 GDA Issues

511 I conclude that the GDA Issue listed in Annex 2 must be satisfactorily addressed before Consent will be granted for the commencement of nuclear island safety-related construction.

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Table 12

Relevant Safety Assessment Principles for Radiological Protection Considered During Step 4

SAP No.	SAP Title	TAG	WENRA Reference*	IAEA Reference**	Contribution of Step 4 Radiological Protection Assessment
Fundamental Principles					
FP.3	Optimisation of protection	T/AST/004	-	SP5 2.2, 2.4 (Ref. 22)	Minor
FP.4	Safety assessment	T/AST/004	-	-	Minor
FP.5	Limitation of risk to individuals	T/AST/004 T/AST/038 T/AST/043 T/AST/045	E1.1	SP6 2.2 (Ref. 22)	Minor
FP.6	Prevention of accidents	T/AST/004 T/AST/045	E2.1	SP8 2.4, 2.5, 2.8 (Ref. 22)	Minor
FP.7	Emergency preparedness and response	T/AST/004	R1.1	SP9 2.5, 2.8 (Ref. 22)	Minor
FP.8	Protection of present and future generations	T/AST/004 T/AST/038	-	SP7 2.2, 2.6 to 2.8 (Ref. 22)	Minor
Radiation Protection Principles					
RP.1	Normal operation	T/AST/038	E1.1	2.4, 4.9 to 4.13, 6.99 to 6.106 (Ref. 22)	Major
RP.2	Accident conditions	T/AST/018 T/AST/038 T/AST/041	E1.1	2.7, 2.8, 4.11 to 4.13 (Ref. 22) 4.40 (Ref. 23)	Major

Table 12

Relevant Safety Assessment Principles for Radiological Protection Considered During Step 4

SAP No.	SAP Title	TAG	WENRA Reference*	IAEA Reference**	Contribution of Step 4 Radiological Protection Assessment
RP.3	Designated areas	T/AST/038	E1.1	6.103 (Ref. 22)	Major
RP.4	Contaminated areas	T/AST/038	E1.1	6.103 (Ref. 22)	Major
RP.5	Decontamination	T/AST/038	E1.1	6.104 (Ref. 22)	Major
RP.6	Shielding	T/AST/002 T/AST/038	E1.1	6.102 (Ref. 22)	Major
Criticality Safety Principles					
ECR.1	Safety measures	T/AST/018 T/AST/041	-	4.80 (Ref. 23)	Major
ECR.2	Double contingency approach	T/AST/041	-	-	Major
Numerical Targets and Legal Limits					
NT.1	Assessment against targets	T/AST/043 T/AST/045	E1.1	-	Major
Target 1	Normal operation – any person on the site	T/AST/043	E1.1	-	Major
Target 2	Normal operation – any group on the site	T/AST/043	E1.1	-	Major
Target 3	Normal operation – any person off the site	T/AST/043	E1.1	-	Major
Target 4	Design basis fault sequences – any person	T/AST/045	E1.1	-	Contribution to fault studies on radiological consequence assessment

Table 12

Relevant Safety Assessment Principles for Radiological Protection Considered During Step 4

SAP No.	SAP Title	TAG	WENRA Reference*	IAEA Reference**	Contribution of Step 4 Radiological Protection Assessment
Target 5	Individual risk of death from on-site accidents – any person on the site	T/AST/045	E1.1	-	Contribution to Level 3 PSA on radiological consequence assessment
Target 6	Frequency dose targets for any single accident – any person on the site	T/AST/045	E1.1	-	Contribution to Level 3 PSA on radiological consequence assessment
Target 7	Individual risk to people off the site from accidents	T/AST/045	E1.1	-	Contribution to Level 3 PSA on radiological consequence assessment
Target 8	Frequency dose targets for accidents on an individual facility – any person off the site	T/AST/045	E1.1	-	Contribution to Level 3 PSA on radiological consequence assessment
Target 9	Numerical targets and legal limits	T/AST/045	E1.1	-	Contribution to Level 3 PSA on radiological consequence assessment
NT.2	Time at risk	T/AST/005 T/AST/043 T/AST/045	E1.1	-	Time of exposure of employees in high dose rate locations

WENRA Reference* refers to the paragraph numbers in Appendix E or Issue R in Ref. 7.

IAEA Reference** refers to the Safety Principles (SAP) in Ref. 27, or to the paragraph numbers in Refs 28 and 29.

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Radiological Protection – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-RP-01	The licensee shall provide a report which demonstrates that the content of cobalt and other elements within primary circuit materials which may become activated and contribute significantly to operator radiation exposure has been reduced so far as is reasonably practicable. The report shall take into account improvements that Westinghouse has identified, in addition to new materials which may have become available following the GDA process.	Mechanical, Electrical and C&I Safety Systems, Structures and Components – delivery to site.
AF-AP1000-RP-02	The licensee shall identify, and provide a justification for, all reasonably foreseeable work activities that are likely to require entry to the Containment whilst at power. For each of these activities, the licensee shall justify the reasons for this exposure, assess likely worker doses, and substantiate whether doses have been reduced so far as is reasonably practicable.	Before fuel on-site.
AF-AP1000-RP-03	The licensee shall, taking into account any changes to the radiation source term and shielding design which have been made since GDA, provide a report that identifies external dose rates for all controlled areas during normal operation. In addition, the licensee shall submit an ALARP justification for areas to which access is required and where the dose rate exceeds 150 micro-sieverts per hour (during normal operation) to demonstrate that dose rates have been reduced so far as is reasonably practicable. The justification shall consider the nature of the work which is likely to be conducted within those areas, including the magnitude and duration of the exposure, and the number of workers exposed.	Before fuel on-site.

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Radiological Protection – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-RP-04	The licensee shall provide a justification for doses incurred during SG lancing, taking into account changes to the radiation source term and shielding design which have been made since GDA. The report shall include an assessment of measures which will reduce worker doses further.	Before fuel on-site.
AF-AP1000-RP-05	The licensee shall carry out an assessment of realistic doses resulting from waste processing and provide a report which substantiates that these doses have been reduced so far as is reasonably practicable.	Before first fuel load.
AF-AP1000-RP-06	The licensee shall provide a report that demonstrates that there is sufficient space in the Fuel Handling Area to load and process a transfer cask in order to allow the despatch of fuel to its chosen design of on-site storage facility, while also restricting worker doses so far as is reasonably practicable.	Before first structural concrete.
AF-AP1000-RP-07	The licensee shall ensure that suitable and sufficient space is available in the design and layout of the site specific health physics facilities (including laboratories, changing/monitoring facilities, emergency facilities and permanent decontamination facilities). The licensee shall provide a justification that the site specific design reduces worker doses, and reduces the likelihood and severity of reasonably foreseeable radiological accidents within the facility, so far as is reasonably practicable. The justification shall be supported by a suitable and sufficient human factors assessment.	Before first structural concrete.

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Radiological Protection – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-RP-08	The licensee shall provide a justification that substantiates that defence in depth has been applied to the contamination control aspects of the AP1000 design, taking into account the response to AF-AP1000-RP-07. In particular, the report shall demonstrate that there are suitable and sufficient barriers to the spread of contamination from controlled areas to non-designated areas. The justification shall be supported by a suitable and sufficient human factors assessment.	Before fuel on-site.
AF-AP1000-RP-09	The licensee shall provide a justification for the exposure of workers in the Fuel Handling Area to radioactive surface and airborne contamination, including substantiation that this exposure has been reduced so far as is reasonably practicable.	Before fuel on-site.
AF-AP1000-RP-10	The licensee shall provide a report that demonstrates that the hierarchy of control measures has been appropriately applied with regard to restricting access to high radiation areas to authorised personnel only.	Before fuel on-site.
AF-AP1000-RP-11	The licensee shall provide a safety case that demonstrates that the on-site specific radiological consequences analyses for design basis events (including hazards) are acceptable and have taken due cognisance of usual UK methodology assumptions, and have explicitly compared the results of those analyses against NT.1 Target 4 regarding the predicted initiating fault frequency versus dose to individuals on the site in the SAPs.	Before fuel on-site.

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Radiological Protection – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-RP-12	The licensee shall provide a safety case that demonstrates that the on-site specific radiological consequences analyses for accidents (including hazards) are acceptable and have taken due cognisance of usual UK methodology assumptions, and have explicitly compared the results of those analyses against NT.1 Target 5 regarding the risk impact to individuals from all the facilities on the site, and against NT.1 Target 6 regarding the predicted single accident frequency versus dose to individuals on the site in the SAPs.	Before fuel on-site.
AF-AP1000-RP-13	The licensee shall provide a safety case to identify the specific areas which are likely to require access during a post-accident phase, and to identify potential doses to workers carrying out those activities and demonstrate that they are ALARP.	Before fuel on-site.
AF-AP1000-RP-14	The licensee shall provide evidence at the construction stage that Metamic™ of the specification used in the safety case is installed in compliance with the design intent.	Before fuel on-site
AF-AP1000-RP-15	The licensee shall establish systems by inactive commissioning to monitor the Metamic™ steel over the lifetime of the plant so as to identify and quantify any degradation.	Before fuel on-site

Annex 2

GDA Issues – Radiological Protection – AP1000

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

SPENT FUEL POOL – CRITICALITY SAFETY CASE

GI-AP1000-RP-01 REVISION 0

<table>
<tr><td>Technical Area</td><td colspan="3">RADIATION PROTECTION</td></tr>
<tr><td>Related Technical Areas</td><td colspan="3">Fault Studies
Radioactive Waste and Decommissioning</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-RP-01</td><td>GDA Issue Action Reference</td><td>GI-AP1000-RP-01.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Westinghouse has not adequately demonstrated why it is not reasonably practicable to design the AP1000 spent fuel pool such that criticality control is achieved through geometrical control and fixed poisons alone.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide a safety case, with supporting evidence, which demonstrates that criticality control of the spent fuel pool is assured for all foreseeable operating conditions through geometrical control and fixed poisons alone.

ONR's expectation is that Westinghouse should adequately apply the hierarchy of safety measures, as described in the HSE's Safety Assessment Principles (SAPs) and international guidance, for criticality control of the AP1000 spent fuel pool. In the context of the design of spent fuel pools at new nuclear power stations, it should be reasonably practicable for Westinghouse to submit an approach that relies on passive safety measures that do not rely on control systems, active safety systems or human intervention.

ONR believes that options to improve the arrangements for spent fuel storage are available to Westinghouse. These options may include, but are not limited to:
Increasing the size of the spent fuel pool.
Redesigning the racking system so that the geometrical separation of fuel assemblies is increased and/or the effectiveness of fixed poisons contained within the racks is improved.
Designing rack inserts containing fixed poisons which can be positioned around fuel assemblies during storage.
Designing fuel assembly inserts to provide fixed poisons.
Utilising additional fuel storage facilities outside the nuclear island to increase storage capacity.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Further explanatory / background information on the GDA Issues for this topic area can be found at:	
GI-AP1000-RP-01 Revision 0	Ref. 82.

Appendix A

Criticality Control of Spent Fuel Storage for the Westinghouse AP1000® Reactor

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Tables

1 INTRODUCTION

1 This Appendix is the report of our work in Step 4 on the topic of the criticality safety of fuel storage in the AP1000. This included a detailed examination of the evidence, on a sampling basis, given by the safety case presented in the GDA submissions.

2 Completion of Step 4 represents the end of our planned GDA assessment on the topic of criticality safety for the Westinghouse AP1000.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR CRITICALITY SAFETY

3 The intended assessment strategy for Step 4 for the criticality safety topic area was set out in an assessment plan that identified the intended scope of the assessment and the standards and criteria that would be applied. This is summarised below.

2.1 Assessment Plan

4 The Assessment Plan for radiological protection of the AP1000 (Ref. 1) identified criticality safety of stored fuel, particularly within the fuel pond, as one of the areas to be assessed during Step 4 of GDA.

2.2 Standards and Criteria

5 The proposal has been compared against HSE's Safety Assessment Principles (SAP) (Ref. 4) specifically covering criticality and engineering.

6 The SAPs set an expectation that the fuel storage design should utilise passive systems to ensure sub-criticality, with a hierarchy based on engineering controls in preference to administrative systems. Geometrical control and fixed poisons are favoured over other means of maintaining criticality control, with an overall aim that the design should eliminate the hazard rather than control it.

7 HSE's SAPs have been benchmarked against international, guidance and I have also taken the opportunity to compare the design against the requirements of two draft IAEA Safety Guides that have been specifically written to address spent fuel storage and criticality safety (Refs A1 and A2). Although still advanced drafts, these guides provide a useful supplement to the SAPs and an indication of up-to-date international experience in this area.

2.3 Assessment Scope

8 Criticality assessment covers the design and intended operational features necessary to prevent the onset of an uncontrolled neutron chain reaction. Typically, a criticality safety case sets out to demonstrate sub-criticality under normal and accident conditions by means of calculations employing computer codes developed specifically for that purpose. The analyses will then aim to demonstrate that the likelihood of the combination of failures necessary for a critical configuration to be achieved is acceptably remote, typically employing the double-contingency principle, supported where appropriate by probabilistic analysis of fault scenarios.

9 This criticality assessment covers the handling of fuel (whether irradiated or not) from receipt and storage of fresh fuel to transfer of used fuel within the fuel pond and subsequent storage. It does not, however, cover emplacement of fuel within the Westinghouse AP1000 vessel nor the arrangement of the elements in the core; these aspects fall within the scope of ND's assessment of Fuel and Core design (Ref. A8). Westinghouse has not put forward a firm proposal for long term storage of spent fuel after it has been removed from the fuel pool, believing this to be an option to be decided by the operator. Instead, Westinghouse has described a dry spent fuel storage option, outside of the nuclear island, based on a Holtec cask system to show that a viable long term storage solution does exist. Dry fuel storage of PWR fuel has not been adopted in the UK but British Energy has submitted a proposal to utilise dry fuel storage at its Sizewell B

plant (Ref. A6) and I was able to see fuel being prepared for dry fuel storage at Farley - a PWR in the US. The proposal has been considered as part of the Radioactive Waste and Decommissioning assessment (Ref. 44), but the design constraints for criticality control are well understood and I have not considered this option in any detail in this assessment.

2.4 Findings from GDA Step 3

10 Criticality assessment of the AP1000 did not start until Step 4 so there are no findings from Step 3. However it was noted in Step 3 (Ref. A7) that:

> *"arguments will be needed to justify why it is not reasonably practicable to enlarge the spent fuel pool to eliminate by design the risk of a criticality fault without the need for administrative controls as would be required by a safety case based upon burn-up credit arguments".*

11 Accordingly it was intended to progress this aspect further within GDA Step 4.

2.4.1 Additional Areas for Step 4 Criticality Safety Assessment

12 None.

2.4.2 Use of Technical Support Contractors

13 GRS was engaged to perform a detailed examination of the content of the safety related submissions to advise whether the proposed criticality safety measures are suitable and sufficient with regard to the "*as low as reasonably practicable*" (ALARP) principle (Ref. 42). The organisation:

- reviewed whether the control measures specified by Westinghouse are suitable and sufficient (and, hence, whether they have taken all reasonably practicable steps to minimise the likelihood of a criticality event);
- reviewed whether the calculation tools, methodologies and assumptions have been appropriately utilised and underpinned by supporting documentation; and
- compared the safety case against SAPs and international guidance.

14 Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH assessed Westinghouse's case for the handling and transfer systems for moving fuel assemblies to and from the reactor pressure vessel.

2.4.3 Cross-cutting Topics

15 Not applicable.

2.4.4 Integration with other Assessment Topics

16 There is a relationship between the capacity of the pond and the cooling time necessary before spent fuel may be transferred to other forms of storage. In my judgement it is likely that Westinghouse can make an adequate safety case, but it may necessitate limiting the capacity of the pond. Current indications are that the capacity may have to be reduced to between ten and twelve years of reactor operation.

17 The design of the racks may also have implications for the pond cooling case.

2.4.5 Out of Scope Items

18 No items have been agreed with Westinghouse as being outside the scope of GDA for criticality safety of fuel storage.

3 WESTINGHOUSE'S SAFETY CASE

19 Westinghouse submitted a design based on the use of racks constructed from stainless steel supplemented by sheets of neutron absorber. Westinghouse has considered a range of neutron absorbers: Boral and Boraflex were rejected in the light of degradation experienced in the USA; borated stainless steel was deemed too brittle to be used in the racks themselves and could incorporate only a limited density of boron; Westinghouse settled on Metamic™, a sintered composite of aluminium and natural boron (which has a composition of approximately 80w% ^{11}B and 20w%^{10}B), which although novel to the UK, has been used successfully in installations (including spent fuel ponds) in the US.

3.1 New Fuel Store

20 A Storage rack is provided in the new fuel storage pit for 72 new assemblies of enrichment up to 5w%. The assemblies are held in a rack which ensures a centre-to-centre spacing of 277mm with Metamic™ sheets to reduce neutron interaction.

21 The pit is normally dry with provision for draining to the radwaste system any water ingress and assemblies can only be emplaced in locations which are designed for that purpose and which do not already contain an assembly.

22 Westinghouse has shown that when dry the reactivity of the system meets the criterion of k_{eff}<0.95 by a substantial margin (Ref. A9), even if fuel assemblies are modelled as having been dropped onto the rack. Westinghouse has analysed the system for the effects of flooding with water of varying density. Full density water was found to provide optimum moderation and K_{eff} was calculated as 0.92077, well within the criterion of K_{eff}<0.95.

3.2 In-containment Storage Rack

23 This rack is provided for temporary storage of fresh or discharged fuel. It comprises a 1x6 array of storage locations on a 277mm pitch with Metamic™ providing neutron absorption.

3.3 Spent Fuel Storage Pond

24 The spent fuel pond was sized for the AP600 and had sufficient capacity to store 619 spent fuel assemblies but the increased power rating of the AP1000 over the AP600 led to Westinghouse exploring an increase in the storage capacity.

25 Increasing the size of the spent fuel pond would create additional room to add storage spaces. However, Westinghouse rejected this option because one of its design objectives for the move from AP600 to AP1000 was not to change the nuclear island footprint or building design. Westinghouse concluded that changing the building design would necessitate a change to the seismic design, which was substantially finished.

26 Westinghouse has described the fuel storage systems in the EDCD (Ref. 60). The Westinghouse fuel pond consists of a concrete structure with free-standing racks designed to maintain a defined separation between fuel elements. When the reactor pressure vessel head is removed e.g. for fuelling, the fuel pond water and primary coolant are able to mix and so the pond water will be dosed with boric acid to protect the reactor system against inadvertent dilution of the boron concentration of the primary circuit.

27 Westinghouse has analysed the stability of Metamic™ against pond parameters and concluded that it is suitable for prolonged use in this environment. These claims are examined elsewhere under the Reactor Chemistry topic area (Ref. 47). Westinghouse's design includes fourteen 'coupons' of Metamic™ which can be periodically removed from the pond for examination to verify the continuing integrity of the material over the lifetime of the plant.

28 The pond is designed to store fuel of maximum enrichment 5w% ^{235}U and utilises two designs of racks:

- Region 1 racks maintain the fuel assemblies on a pitch of 278mm and are designed to be sub-critical for fuel irrespective of its irradiation, including unirradiated fuel.
- Region 2 racks maintain a pitch of 230mm and are designed for storage of irradiated fuel which complies with design limits on the combination of initial enrichment and irradiation. E.g. fuel of initial enrichment of 5w% ^{235}U would require a minimum rradiation of 40Gwd/tU.

29 The locations formed by the Region 1 and Region 2 racks are on pitches of 278mm and 230mm respectively. Criticality control is achieved by the geometrical spacing defined by the racking system together with the neutron absorption properties of the boron in the Metamic™.

30 Neutrons leaving each assembly are slowed (thermalised) by the water surrounding the element, and since ^{10}B has a high neutron capture cross section in the thermal region this increases the efficiency of the Metamic™ in absorbing the neutrons. The size of the water gap therefore plays an important part in helping to limit the reactivity of the racks.

31 Westinghouse has conservatively modelled Region 1 racks with unirradiated fuel elements of 5w% ^{235}U. The Region 1 racks were modelled as an infinite array and shown by calculation to meet the criteria of Keff <0.95 for normal conditions when filled with unirradiated fuel in pure water. No claim is made on the presence of soluble boron.

32 The Region 2 racks are more closely spaced, reducing the amount of water between fuel assemblies and so also reducing the effectiveness of the Metamic™.

33 For the analysis of the Region 2 racks Westinghouse devised a model for the fuel specification: this is intended to be a conservative representation of fuel with initial enrichment 5w% ^{235}U irradiated to approximately 40Gwd/t. Irradiated PWR fuel is typically less reactive than fresh fuel, which makes it easier to satisfy the relevant criteria for the deterministic calculations, but does place reliance on ensuring only sufficiently irradiated fuel is stored in these locations.

34 Westinghouse proposed that fuel in Region 2 racks would be managed in accordance with a loading curve which traced enrichment against irradiation. Also, Westinghouse proposed that the control of the emplacement of fuel would be managed by a computer code that pre-programs the movement of fuel assemblies to specific locations. This would offer protection against placing fuel which fell outside the loading curve into the Region 2 racks.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR CRITICALITY SAFETY

35 ND expects that criticality safety will be controlled through engineering features built into the design. For fuel storage at its simplest, this may take the form of geometrical constraints to maintain sufficient separation between fuel elements to prevent significant nucleonic interaction: typically 300mm would be sufficient. However, the Westinghouse proposal seeks to maximise the number of fuel assemblies accommodated in a given space, while minimising the risk of a criticality excursion and ensuring adequate cooling. This inevitably makes the safety case more complex and it is on this aspect that my assessment has primarily focused.

4.1 New Fuel Storage Rack

4.1.1 Assessment

36 The new fuel storage rack is intended to be dry but is in any case shown to be sub-critical under flooded conditions and tolerant against foreseeable fault scenarios. This is achieved inherently through geometrical spacing of the fuel storage positions, plus the addition of Metamic™ as a fixed poison.

4.1.2 Findings

37 My assessment has found Westinghouse's safety case for the new fuel rack to be acceptable. However the Metamic™ plays an important role in ensuring safety and the material's compliance with the specification implied in the safety case should therefore be verified before operation commences; this will be pursued through Assessment Finding **AF-AP1000-RP-14**:

> ***AF-AP1000-RP-14:*** *The licensee shall provide evidence at the construction stage that Metamic™ of the specification used in the safety case is installed in compliance with the design intent.* Milestone: Before-on-site fuel.

4.2 In-containment Storage Rack

4.2.1 Assessment

38 This rack maintains the fuel assemblies on a similar pitch to the Region 1 racks in the spent fuel pond, but because they are in the form of a 1 x 6 array, neutron leakage is much greater. Therefore the in-containment storage racks are shown to be sub-critical by a substantial margin when filled with fresh fuel at 5w% ^{235}U enrichment and pure water. This is achieved through geometrical control, particularly with the greater neutron leakage afforded by the 1 x 6 array and the addition of Metamic™ as a fixed poison. The intended presence of soluble boron has not been taken into account in the analysis and so provides a significant additional safety factor.

4.2.2 Findings

39 My assessment has found Westinghouse's safety case for the in-containment rack to be acceptable. However the Metamic™ plays an important role in ensuring safety the material's compliance with the specification implied in the safety case should therefore be verified before operation commences. This will be pursued through Assessment Finding **AF-AP1000-RP-14** above.

4.3 Spent Fuel Pond

4.3.1 Assessment

40 Taking credit for the irradiation history of fuel (burn-up credit) is an established practice in the USA where a number of fuel ponds have been re-racked to increase the utilisation of existing pond space by reducing the separation between fuel assemblies.

41 Because the UK has historically reprocessed spent fuel there has not been the same pressure on storage facility capacity that has developed in some other countries where fuel has been stored for later disposal. In addition Magnox and AGR fuel are optimised neutronically for use in CO_2 and are less reactive in water than PWR fuel. The driver to take credit for burnup has therefore not developed to the same extent in the UK for spent fuel storage; but instead the reduction in reactivity due to irradiation has been regarded as an unquantified, but significant, additional safety factor.

42 Modelling of burnup is complex because during irradiation some fissile nuclides, notably ^{235}U are reduced but others, e.g. ^{239}Pu, are created. Other nuclides may be created and then through decay or interaction are transformed again into different nuclides. The exact composition of the fuel after a period of irradiation may be influenced by the position in the reactor, the operating history and soluble boron levels and the extent to which control rods were used. Even then the irradiation is not uniform along the length of the assembly and some form of approximation is necessary to derive a model which bounds all fuel. These complexities make burnup credit arguments quite complicated in criticality safety cases and lend weight to the UK practice of simply regarding burnup as an additional unquantified safety factor.

43 Additionally, however, there is no UK experience of implementing burnup credit for fuel storage and even in the USA where burnup credit are used for some safety cases the NRC had indicated to industry that it intended to review its own guidance on the use of burnup credit.

44 The safety case presented in the European Design Control Document (EDCD), Ref. 60, for the Region 2 racks has two distinct but linked aspects of particular interest:

- The calculational methodology Westinghouse has employed to represent the effects on irradiation of fuel elements; and
- The operational controls invoked to assure that fuel elements outside the allowable irradiation limit are not placed in Region 2 racks.

45 Before the criticality assessment started, I perceived that the regulatory assessment was unlikely to be acceptable for GDA because:

- Burnup credit methodology has not to date been employed in the UK;
- Only limited experimental validation work was apparently available to Westinghouse's designers to support the burnup credit analysis;
- There was a high degree of reliance on administrative/software control to prevent the misplacing of fuel assemblies

46 I explained to Westinghouse in July 2008 that their reliance on burnup credit represented a programme risk and that it would be challenging for Westinghouse to present arguments to substantiate why reliance on administrative controls would be ALARP.

47 On commencement of the criticality assessment I raised RO-AP1000-73 (Ref. 9) asking Westinghouse to demonstrate that:

- the chosen design was ALARP;
- the calculations were based on appropriate validation; and
- all appropriate accident scenarios had been considered.

48 Westinghouse responded in August 2010 by reviewing four options:

i) The burnup credit case described above;

ii) Replacing the Region 2 racks with Region 1 racks;

iii) The same racking arrangements but blanking 2 out of every 4 locations of the Region 2 racks;

iv) The same racking arrangements but blanking 1 out of every 4 locations of the Region 2 racks.

49 Westinghouse discounted option (i) because of the reservations I had expressed. Of the remaining three, Westinghouse favoured option (iv) believing it would be consistent with ND's SAPs and would give more storage capacity than the other options.

50 Westinghouse's case for option (iv) shows that the calculated neutron multiplication factor (K_{eff}) for all positions containing unirradiated fuel of maximum enrichment (5w% ^{235}U) satisfies the criteria of k_{eff}< 0.95 but only provided that the soluble boron concentration in the spent fuel pond water remains at least 1300ppm.

51 Westinghouse identified a number of fault scenarios which are consistent with experiences of pond storage. Westinghouse's analysis of a fuel assembly dropped onto the top of the fuel racks shows that the deformation of the racks would be minimal and would not compromise the neutron absorption properties of the Metamic™. Further analysis shows that the K_{eff} in this fault scenario would not exceed 0.98 provided that the boron content of the pond water is maintained at or above 1000ppm.

52 Westinghouse's calculations have been performed with the Monte Carlo neutronics code Keno Va and suitable recognised and validated datasets. The fuel and racks have been modelled with due regard to conservative representation of manufacturing tolerances such as fuel pellet density and Metamic™ thickness and with full density water, shown by scoping calculations to be the worst case.

53 Westinghouse's deterministic calculations are methodical and well presented and the relevant faults have been identified and considered. There is sufficient relevant verification of codes employed. The design put forward has been accepted by the United States Regulatory Commission (US NRC) as it was supported by a safety case following US NRC guidance – including the use of burnup credit.

54 GRS ran independent criticality calculations using the Monte Carlo code KENO and found that Westinghouse's calculations were reproducible and the methodology appropriate to the systems being modelled – i.e. dry and wet systems containing low enriched uranium dioxide as well as boron.

55 GRS concluded that for the AP1000 (subject to a reservation on credit for soluble boron in the fuel pond - see below) the design was appropriate to maintain sub-criticality under normal and accident conditions.

56 However, in my judgement the case submitted in response to RO-AP1000-073 does not adequately show that option (iv) is an ALARP solution. In particular, although the case did not explicitly rely on burnup credit, it did require a concentration of 1300ppm of boron to ensure sub-criticality under non-accident conditions. This is not consistent with relevant good practice nor IAEA draft guidance (Ref. A1) which recommends that:

> *"the presence of a soluble neutron absorber in the storage pond water should not be taken into account in the criticality safety demonstration for normal operation."*

57 In addition GRS noted that the proposed reliance on boron under normal conditions :

> *'is not fully in compliance with basic requirements and current practice in criticality safety. It is recommended to HSE-ND not to accept this approach'*

58 I concluded that the Westinghouse design is not acceptable because:

- Geometrical constraint is not the principle means of criticality control (SAPs para. 471).
- The opportunity has not been taken to design out the risk of criticality, rather the design attempts to control it (SAP para 136).
- It is inconsistent with the double contingency principle (SAPs para. 474) which underpins ND's expectations for criticality control.
- It does not align with the principle on the hierarchy of safety measures (SAPs para. 146).

59 While there are strong arguments why the loss of boron is unlikely, I concluded that a new design should not have to rely on soluble poisons to ensure sub-criticality, but rather that it should be possible to control criticality through geometrical and fixed poisons alone. In view of the importance of this finding to the design I sought peer review within a forum of other ND criticality practitioners, and they endorsed my conclusion. The peer group concluded that the use of fixed poisons and geometrical control represented relevant good practice in that it had been the design basis for many existing facilities and there seemed no good reason why it should not be the basis for a new plant.

60 Following discussions of this conclusion between ND and Westinghouse in Pittsburgh, ND wrote (Ref. A3) to clarify the position and request that Westinghouse should re-evaluate the options for spent fuel storage and present a solution that achieves criticality control through geometrical control and fixed poisons alone. The letter pointed out that the criticality case had the potential to become a GDA Issue.

61 Westinghouse responded (Ref. A4) with a proposal for a variant of the 3 out of 4 option discussed above, but it relied on credit being taken for burnup.

62 In response to my requests for a review of options and a revised safety case, a meeting was convened in Manchester in February 2011 to allow Westinghouse to present the early outcomes of their ALARP study. Westinghouse explained that they had designed the pond to have storage capacity for 15 years and that a design which provided less than 10 years capacity would not in their view be commercially viable.

63 Westinghouse reviewed options (ii) – (iv) already discussed above, and in addition considered:

- Enlarging the pond.
- Building an additional pond.
- Option (iv) above plus inserting poisoning devices into fuel assemblies.

64 Westinghouse staff emphasised that in their view it was not reasonably practicable to enlarge the pond because of civil structure and seismic requirements, meaning that any solution had to be within the constraint of the existing pond size.

65 Equally Westinghouse rejected the building of an additional pond as it considered this would involve disproportionate cost and an additional campaign of fuel movement..

66 The remaining potential options Westinghouse identified are summarised in Table 2.

67 Preliminary calculations by Westinghouse indicated that it might be possible to demonstrate safety with the benefit of poison inserts without the need to take credit for burnup or soluble boron, although these parameters would inevitably provide further margins of safety. But ND would need to be convinced that this solution provides as effective a safety margin as the geometrical control solutions i.e. 2 out of 4 or all replacing Region 2 racks with Region 1.

68 Enlarging the pond would be the most elegant solution from a safety perspective, allowing the capacity to be maintained while using lower density racking. Should pond enlargement be proven not to be reasonably practicable then reducing the storage density would allow more flexibility in the safety case.

69 Of the options considered to date, the use of all Region 1 racks would also be acceptable on criticality safety grounds. It would allow the use of all racks for fuel of any irradiation history without reliance on soluble boron, so meeting ND's expectations of relevant good practice.

70 It may be that other solutions could provide comparable levels of safety in which case they would be acceptable – of those under present consideration options (iii) and poisoned inserts used in conjunction with option (iv) would appear to have possibilities.

71 Table A2 presents a summary of the options discussed and the calculated k_{eff} under a range of conditions. Since no solution was identified which would obviously satisfy both Westinghouse and ND I raised GDA Issue **GI-AP1000-RP-01** requiring Westinghouse to:

> ***GI-AP1000-RP-01:*** *Provide a safety case, with supporting evidence, which demonstrates that criticality control of the spent fuel pool is assured for all foreseeable operating conditions through geometrical control and fixed poisons alone.*

72 The complete GDA Issue and associated action is formally defined in Annex 2 to the main text of this report.

73 I expect Westinghouse's resolution plan to include an ALARP review of the available options and provision of a safety case to demonstrate that the chosen option meets relevant good practice.

4.3.2 Findings

74 The design of the Region 1 racks allows fuel of any irradiation to be stored in an arrangement which will be sub-critical under normal and foreseeable fault conditions. But Westinghouse has not yet made an acceptable safety case for the design of the fuel storage pond because the design of the Region 2 racks has not been demonstrated to be sub-critical under normal conditions without credit being taken for soluble boron. As a result I have raised a GDA Issue as follows:

> ***GI-AP1000-RP-01:*** *Provide a safety case, with supporting evidence, which demonstrates that criticality control of the spent fuel pool is assured for all foreseeable operating conditions through geometrical control and fixed poisons alone.*

75 All the permutations under consideration utilise Metamic™ and it is necessary to ensure that the specified material is installed at construction and that procedures are in place to monitor any degradation over the lifetime of the plant. This will be pursued through Assessment Findings **AF-AP1000-RP-14** (above) and **AF-AP1000-RP-15**.

> ***AF-AP1000-RP-15:*** *The licensee shall establish systems by inactive commissioning to monitor the Metamic™ steel over the lifetime of the plant so as to identify and quantify any degradation.* Milestone: Before fuel on-site.

4.4 Overseas Regulatory Interface

76 I had some discussions with NRC at an early stage of the assessment on their guidance for burnup credit methodology which I took into account during my assessment.

4.5 Interface with Other Regulators

77 Any reduction in cooling time in the pond brought about by restrictions on the capacity might have implications for disposability. I have therefore kept the Environment Agency appraised of discussions and developments on the assessment of the spent fuel pond.

4.6 Other Health and Safety Legislation

78 The Ionising Radiations Regulations 1999 and the Construction Design and Management Regulations are both relevant to criticality safety of the design of the Westinghouse AP1000. Consideration of these has been included in my assessment where appropriate.

5 CONCLUSIONS

79 This is the Step 4 report on HSE's GDA work for the Westinghouse AP1000 criticality safety assessment.

80 The safety cases for the new fuel storage rack and the in-containment fuel storage rack are based on geometrical control and the use of fixed poisons and have been found to be adequate.

81 I am not satisfied that the claims, arguments and evidence laid down within the Consolidated GDA Safety Submissions for the criticality safety of the spent fuel pond represents an ALARP position. A design in which sub-criticality under normal conditions is dependent upon maintenance of the boron concentration in the fuel pond is unsatisfactory as it is not aligned with the SAPs, IAEA guidance nor relevant good practice.

5.1 Key Findings from the Step 4 Assessment

82 Westinghouse has already identified and analysed options which could be satisfactory from a criticality safety perspective. These include:

- Replacing Region 2 racks with Region 1 racks.
- Blocking off 2 out of every 4 channels in Region 2 racks.
- The use of poisoned inserts in conjunction with the blocking of 1 out of 4 channels in Region 2 racks.

83 The presence of Metamic™ is an important factor in maintaining sub-criticality and its specification and continuing presence should be verified at construction and during operation.

5.1.1 Assessment Findings

84 I conclude that the following Assessment Findings (listed in Annex 1 of the main report) should be programmed during the forward programme for Westinghouse AP1000 as normal regulatory business:

> ***AF-AP1000-RP-14:*** *The licensee shall provide evidence at the construction stage that Metamic™ of the specification used in the safety case is installed in compliance with the design intent.* Milestone: Before fuel on-site.
>
> ***AF-AP1000-RP-15:*** *The licensee shall establish systems by inactive commissioning to monitor the Metamic™ steel over the lifetime of the plant so as to identify and quantify any degradation.* Milestone: Before fuel on-site.

5.1.2 GDA Issues

85 I conclude that GDA Issue **GI-AP1000-RP-01** listed in Annex 2 of the main body of this report must be satisfactorily addressed before Consent should be granted for the commencement of nuclear island safety-related construction.

6 REFERENCES

A1 *Storage of Spent Fuel.* International Atomic Energy Agency (IAEA). Draft Safety Guide DS371. IAEA. 11 November 2008. TRIM Ref. 2011/565547.

A2 *Criticality Safety for Facilities and Activities Handling Fissionable Material. .* International Atomic Energy Agency (IAEA). Draft General Safety Guide GSG DS405 Version 4. IAEA. 10 December 2010. TRIM Ref. 2011/565535.

A3 *Spent Fuel Pool – Criticality Safety Case.* ND Letter UN WEC70261R. 30 November 2010. TRIM Ref. 2010/565480.

A4 *Response to WEC70261R - Spent Fuel Pool – Criticality Safety Case.* Westinghouse letter UN REG WEC00478. 14 January 2011. TRIM Ref. 2011/94667.

A5 Not used.

A6 *Sizewell B Dry Fuel Store - Application to the Secretary of State for the Department of Energy and Climate Change (Electricity Act 1989 Section 36).* February 2010.

A7 *Step 3 Fault Studies Assessment of the Westinghouse AP1000*. HSE-ND Assessment Report AR 09/018. November 2009. TRIM Ref. 2009/335824.

A8 *Step 4 Fuel and Core Design Assessment of the Westinghouse AP1000® Reactor*. ONR Assessment Report ONR-GDA-AR-11-005, Revision 0. TRIM Ref. 2010/581526.

A9 *AP1000 Standard Combined License Technical Report - New Fuel Storage Rack Criticality Analysis.* APP-GW-GLR-030 Revision 0. Westinghouse Electric Company LLC. May 2006. TRIM Ref. 2011/576548.

Table A1

Relevant Safety Assessment Principles for Criticality Safety Considered During Step 4

SAP No.	SAP Title	Description
EKP.	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.5	Safety measures	Safety measures should be identified to deliver the required safety function(s).
ECR.1	Criticality safety	Wherever significant amount of fissile materials may be present, there should be a system of safety measures to minimise the likelihood of unplanned criticality.
ECR.2	Double contingency approach	A criticality safety case should incorporate the double contingency approach.

Table A2

Summary of Options Presented by Westinghouse During GDA

No.	Configuration	Irradiation Credit	Other	Boron Concentration (ppm) to meet safety criteria under Normal Conditions	K_{eff} Criteria	Boron Concentration (ppm) to meet safety criteria under Accident Conditions		Storage Capacity (Years)
						Boron (ppm)	K_{eff}	
1	Original Case	e.g 30Gwd/t – 5%U235		500 0	<0.95 <1.0	963	0.95	15
2	All Region 1	Nil		0	<0.95	"Low"	<0.98	10
3	3 out of 4	Nil		1300	<0.95	1000<0.98		12
4				1000	<0.98			
5	2 out of 4	Nil		0	<0.95	250	<0.98	9
6	Jan 2011 Case 3 out of 4 – Region 2 fuel irradiated through 1 cycle	20Md/t		0	<0.95			12