Office for Nuclear Regulation

An agency of HSE

Generic Design Assessment – New Civil Reactor Build

Step 4 Radiological Protection Assessment of the EDF and AREVA UK EPR™ 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 EDF and AREVA relating to the UK EPR™ 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 EDF and AREVA 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 EDF and AREVA UK EPR™ reactor.

EXECUTIVE SUMMARY

This report presents the findings of the Radiological Protection assessment of the UK EPR™ reactor undertaken as part of Step 4 of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA). The assessment has been carried out on the November 2009 Pre-construction Safety Report (PCSR) and supporting documentation submitted by EDF and AREVA during Step 4.

This assessment has followed a step-wise-approach in a claims-argument-evidence hierarchy. In Step 3 the claims made by EDF and AREVA in particular topic areas of Radiological Protection 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 UK EPR 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 PCSR 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 the Radiological Protection an assessment plan for Step 4 was set-out in advance.

My assessment has focused on:

- Engineered features that would restrict exposures of workers to ionising radiation so far as is reasonably practicable during normal operation.
- The approach taken to optimise exposures of workers to ionising radiation when carrying out high dose work activities.
- Engineered features would restrict exposures of workers to ionising radiation so far as is reasonably practicable during accident conditions.

A number of items have been agreed with EDF and AREVA as being outside the scope of the GDA process and hence have not been included in my assessment, such as operational feedback data from operating nuclear power plants, and the selection of operating equipment and comparison of existing suppliers.

From my assessment, I have concluded that:

- The plant and its operations have been designed to ensure that engineered features would restrict exposures to workers to ionising radiation so far as is reasonably practicable during normal operation.
- There was a suitable and sufficient systematic and comprehensive approach to optimising exposures of workers to ionising radiation when carrying out high dose work activities.
- The plant and its operations have been designed to ensure that engineered features would restrict exposures to workers to ionising radiation so far as is reasonably practicable during accident conditions.

In some areas there has been a lack of detailed information which has limited the extent of my assessment. As a result HSE's Nuclear Directorate (ND) 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.

Some of the observations identified within this report are of particular significance and will require resolution before HSE would agree to the commencement of nuclear safety-related construction of a UK EPR reactor in the UK. This is identified in this report as a GDA Issue and is contained in Annex 2. In summary this relates to:

- Radiological zoning for restriction of exposure to ionising radiation of workers is fundamental to the design of the nuclear island, and bulk shielding is inextricably linked with civil engineering aspects of that design. The radiological zoning classification scheme underpinned by design shielding calculations is not referenced in the GDA submission for the UK EPR design.
- To resolve this GDA issue, EDF and AREVA are required to provide an overview document that supplements the claims and arguments presented in the PCSR Chapter 12.3 with additional information on the radiological zoning classification scheme for the nuclear island, including dose rate criteria ad predictions for all modes of plant operation, for occupied areas as a direct reference from the PCSR.

Overall, based on the sample undertaken in accordance with ND procedures, I am broadly satisfied that the claims, arguments and evidence laid down within the PCSR and supporting documentation submitted as part of the GDA process present an adequate safety case for the generic UK EPR reactor design. The UK EPR reactor is therefore suitable for construction in the UK, subject to satisfactory progression and resolution of a GDA Issue 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.

LIST OF ABBREVIATIONS

ACOP	Approved Code Of Practice
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)
cPCSR	Consolidated Pre-construction Safety Report
DAC	Design Acceptance Confirmation
DECC	Department for Energy and Climate Change
EDF and AREVA	Electricité de France SA and AREVA NP SAS
EDPI	Initial Predicted Dose Estimate
EDPO	Optimised Predicted Dose Estimate
EPR10	Environmental Permitting Regulations 2010
FA3 EPR	Flamanville 3 EPR (in France)
GDA	Generic Design Assessment
GRS	Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) 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
KRC	Body Contamination and Dosimetry Control System
KRT	Plant Radiation Monitoring System (PRMS)
IAEA	The International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IRR99	Ionising Radiations Regulations 1999
IRSN	The French Nuclear Safety Authority's Technical Support Organisation
IRS	International Reporting System
ISIO	In-Service Inspection Outage
MCNP	Monte Carlo N-Particle (A Shielding Code)
MDEP	Multinational Design Evaluation Programme
MHSWR99	Management of Health and Safety at Work Regulations 1999, as amended

LIST OF ABBREVIATIONS

ND	The (HSE) Nuclear Directorate
NDT	Non-Destructive Testing
NEA	Nuclear Energy Agency
NIA65	Nuclear Installations Act 1965, as amended
NPP	Nuclear Power Plant
NRO	Normal Refuelling Outage
NT	Nuclear Technologies
NRPB	National Radiological Protection Board (now the HPA-CRCE)
OECD	Organisation for Economic Co-operation and Development
PCER	Pre-construction Environment Report
PCSR	Pre-construction Safety Report
POCO	Post Operational Clean Out
PPE	Personal Protective Equipment
PRMS	Plant Radiation Monitoring System
PSA	Probabilistic Safety Analysis
PWR	Pressurised Water Reactor
REACT	REACT Engineering Limited
REPPIR	Radiation (Emergency Preparedness and Public Information) Regulations 2001
RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
ROA	Regulatory Observation Action
ROO	Refuelling Only Outage
RPE	Respiratory Protective Equipment
SAP	Safety Assessment Principles
SFAIRP	So Far As Is Reasonably Practicable
SG	Steam Generator
SSM	The Swedish Nuclear Safety Authority
STUK	The Finish Nuclear Safety Authority
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query
TSC	Technical Support Contractor

LIST OF ABBREVIATIONS

TUV SUD	TÜV SÜD Industrie Service GmbH
US NRC	Nuclear Regulatory Commission (United States of America)
WENRA	Western European Nuclear Regulators' Association

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1 INTRODUCTION

1 This report presents the findings of the Step 4 Radiological Protection assessment of the November 2009 UK EPR Pre-construction Safety Report (PCSR) (Ref. 11), and supporting documentation provided by EDF and AREVA under the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process. Assessment was undertaken of the PCSR (Ref. 11) and the supporting evidentiary information derived from the Submission Master List (Ref. 12). The approach taken was to assess the principal submission, i.e. the PCSR, and then undertake assessment of the relevant documentation sourced from the Submission Master List on a sampling basis in accordance with the requirements of 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 EDF and AREVA 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 The November 2009 PCSR (Ref. 11) was revised to take account of information / documentation requested and assessed by ND, and this consolidated PCSR (cPCSR) (Ref. 13) was submitted in March 2011 with the revised Submission Master List (Ref. 16). My report also considers updates to the cPCSR (Ref. 13) that relate to outcomes following completion of EDF and AREVA's dose optimisation studies.

4 A number of items have been agreed with EDF and AREVA 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

5 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

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

2.2 Standards and Criteria

7 The key piece of legislation for nuclear facilities is the Nuclear Installations Act 1965, as amended (NIA65) (Ref. 20). The standards and criteria that were used to judge radiological protection in the UK EPR are legislation, SAPs (Ref. 4) and Technical Assessment Guides (TAG). The SAPs (Ref. 4) and TAGs were developed by ND, and the SAPs (Ref. 4) are published by HSE. Another key piece of legislation is the Ionising Radiations Regulations (IRR99) (Ref. 17). These Regulations implement the European Basic Safety Standards Directive (Ref. 15), which in turn takes into account recommendations from the International Commission on Radiological Protection (Ref. 14). 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. 18), Management of Health and Safety at Work Regulations 1999, as amended (MHSWR99) (Ref. 125) and Environmental Permitting Regulations 2010 (EPR10) (Ref. 19).

8 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" (SFAIRP) 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 SFAIRP.
- Exposures to ionising radiation should be limited in that they must not exceed the statutory dose limits in IRR99 (Ref. 17). Clearly this should not be an issue for modern nuclear plant under normal operation.

9 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 covering optimisation of protection, safety assessment, limitation of risks to individuals, prevention of accidents, emergency preparedness and response, and protection of present and future generations. 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 my assessment. The Section of the SAPs on Radiation Protection (Ref. 4) also refers to IRR99 (Ref. 17), and in particular to the Approved Code of Practice (ACOP) and guidance to IRR99 on the hierarchy of control measures in regulation 8 (Ref. 21). The criticality safety principles (ECR.1 and ECR.2) are for safety measures and double contingency approach, and these areas were covered by my assessment.

10 All the numerical targets and legal limits (NT.1 Targets 1 to 9 and NT.2) are relevant to some 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 Analysis (PSA) was taken by ND assessors in other disciplines, and 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 PSA assessment report for the UK EPR, Ref. 46). These principles, targets and limits were assessed on a sampling basis. They will also need to be considered during the site-specific assessment phase (Phase 2).

11 IRR99 (Ref. 17) require that, in general, the annual dose limit for workers is $20mSvy^{-1}$. Although the annual dose limit applies to all employees on the site, ND's SAPs (Ref. 4) provide three values for the Basic Safety Levels (BSL) (the levels of dose above which the risk of harm is considered intolerable) for employees with different roles. The BSL for individual workers who are working with ionising radiation during normal operation (NT.1 Target 1), is the same value as the annual dose limit under IRR99 (Ref. 17), namely $20mSvy^{-1}$. The BSL for individuals within a group of workers who are working with ionising radiation during normal operation is half of that value, namely, $10mSvy^{-1}$ (NT.1 Target 2). The BSL for other persons on-site during normal operation (e.g. workers not working with ionising radiation, visitors) is $2mSvy^{-1}$ (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. 17), namely $1mSvy^{-1}$ (NT.1 Target 3).

12 ND's Basic Safety Objective (BSO) is the level below which it would not be reasonable to use 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. 17), namely $1mSvy^{-1}$ (NT.1 Target 1). The BSO for individuals within a group of workers who are working with ionising radiation during normal operation is also one twentieth of the BSL, namely, $0.5mSvy^{-1}$ (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 $0.1mSvy^{-1}$ (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. 17), namely $0.02mSvy^{-1}$ (NT.1 Target 3).

13 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, whereas 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) which 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.

14 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.

15 The TAGs of most relevance to the assessment are on fundamental principles (Ref. 31), demonstration of ALARP (Ref. 32), radiological protection (Ref. 33), shielding (Ref. 29), criticality safety (Ref. 44), criticality warning systems (Ref. 45), radiological analysis during normal operation (Ref. 34), and radiological analysis during fault conditions (Ref. 35).

16 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 1, along with relevant Western European Nuclear Regulators' Association (WENRA) and International Atomic Energy Agency (IAEA) references (Ref. 7, and Refs 27, 28 and 42, respectively) and TAGs (Refs 31, 32, 33, 29, 44, 45, 34 and 35). 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 (NEA) of the Organisation for Economic Co-operation and Development (OECD) has also published guidance on occupational radiological protection principles and criteria for designing new nuclear power plants (Ref. 43). This guidance includes the new EPR as a case study.

17 The principal standards and criteria for judging whether ALARP has been met are the ACOP and guidance to IRR99 (Ref. 21), supplemented by additional guidance on HSE's website (including the TAGs). In addition, IRR99 (Ref. 17) 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.

18 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. 22), supplemented by additional guidance on HSE's website (Ref. 23). The National Radiological Protection Board (NRPB), now the Centre for Radiation, Chemical and Environmental Hazards of the Health Protection Agency (HPA-CRCE), has published guidance on controlling doses for people on-site during radiation accidents (Ref. 24).

19 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 (HSWA74) (Ref. 25), refer to risks, as do the expectations in many of the SAPs (Ref. 4). However, the duties of radiation employers in IRR99 (Ref. 17) and expectations 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. 17) is also applicable here, as the 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. 21). 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.

20 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. 17) is enforced by ND on behalf of HSE, and EPR10 (Ref. 19) is enforced by the Environment Agency. IRR99 (Ref. 17) requires dose constraints to restrict exposure to ionising radiation at the planning stage where it is appropriate to do so. The guidance to IRR99 (Ref. 21) advises that a constraint for a single new source should not exceed $0.3mSvy^{-1}$ 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 duty holder's control, since this is an entity for which radiological protection can be optimised as a whole. However, HPA-CRCE has recently recommended that doses to members of the public from new nuclear power plants (NPP) should be constrained to $0.15mSvy^{-1}$ (Ref. 26).

21 The ALARP principle also applies to manufacturers etc. Section 6 of HSWA74 (Ref. 25) 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. 17). 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 SFAIRP 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

22 The objective of the Step 4 assessment was to review the safety aspects of the proposed reactor designs in a more detailed way by examining the evidence, supporting arguments and claims made in EDF and AREVA's 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.

23 The Step 4 assessment dealt with occupational and public doses being ALARP during normal operation. This assessment re-visited HSE's Step 3 assessment in light of detailed evidence submitted by EDF and AREVA during Step 4 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, plant radiation monitoring system (PRMS), and decommissioning. The assessment initially focused on high dose work activities, and particularly on dose optimisation for jumpers (workers whose work activities are performed in locations with high dose rates, and whose exposure times are measured in seconds). However, during Step 4, operation and maintenance activities involving jumpers during mid-loop operations with regard to fitting of steam generator (SG) nozzle dams were agreed with EDF and AREVA as being out of scope of GDA

since the UK EPR design does not require these activities to be undertaken (this would be an operational option).

24 In addition, the assessment included occupational and public radiation exposure during accident conditions, where occupational radiation exposure included criticality accidents. Other assessors looked at accident risk, and the Radiological Protection assessment contributed to the Step 4 analysis of Level 3 PSA with regard to plume dispersion modelling and dose consequences. As already noted, the Level 3 PSA assessment is reported in the Step 4 PSA assessment report for the UK EPR (Ref. 46).

25 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), criticality, reactor chemistry, radioactive waste management, decommissioning, mechanical engineering, human factors, public exposures, and control and instrumentation, as necessary.

26 Regulation of public radiation exposure is shared between the Environment Agency (in England and Wales) and HSE. 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. 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. In contrast to this, for public doses resulting from accidents, ND leads for all pathways (i.e. doses resulting not only from direct radiation originating from within the site boundary, but also from radioactive substances escaping into the environment).

27 A number of other topic areas in the SAPs (Ref. 4) which have some relevance to radiological protection include 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.

28 The overall bases for the start of assessment in GDA Step 4 were the radiological protection elements of the following.

- The update to the Submission / PCSR / Supporting Documentation, and the Design Reference Point that relates to the Submission / PCSR as set out in the UK EPR GDA Project Instruction UKEPR-1-002 (Ref. 47): these submissions should fulfil the requirements of the GDA guidance to Requesting Parties (Ref. 48).
- Design Change Submissions – which are proposed by EDF and AREVA and can be incorporated within the GDA scope by agreement with the Assessment Unit Heads.

29 The conclusions of the Radiological Protection assessment for GDA Step 4 included verifying that all matters that had been resolved were suitably dealt with in the Submission / Consolidated PCSR.

2.3.1 Findings from GDA Step 3

30 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 in Phase 2 by the licensee. In such cases, the design would need to be sympathetic to the needs of the licensee (e.g. allowing sufficient space to allow erection of temporary shielding in locations where provision of permanent shielding may not be ALARP).

31 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) or Regulatory Issues (RI) were identified. Initially there was the potential for a RO to be raised on optimisation for "jumpers" during Step 4, although as previously noted, operation and maintenance activities involving jumpers during mid-loop operations with regard to fitting of steam generator nozzle dams were agreed with EDF and AREVA during Step 4 as being out of scope of GDA.

32 The Step 3 assessment report (Ref. 6) concluded that EDF and AREVA 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

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

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

- Radiation sources.
- Designated areas (radiological classification of areas / radiological zoning).
- Shielding.
- Contaminated Areas.
- Ventilation.
- PRMS.
- Decontamination.
- Optimisation for work activities (including fuel route).
- Optimisation for jumpers (maintenance activities involving jumpers during mid-loop operations with regard to fitting of SG nozzle dams – this was subsequently agreed with EDF and AREVA as being outside the scope of GDA).
- Waste handling and decommissioning.
- Public exposure from direct shine (direct radiation originating from within the site boundary).

35 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 report of the UK EPR (Ref. 46).

2.3.3 Use of Technical Support Contractors

36 A Technical Support Contractor (TSC), AMEC, was engaged to assist in the topic areas of radiological protection and radioactive waste and decommissioning during Steps 3 and 4. The TSC's task was to prepare a report of a literature review of radiological protection and radioactive waste management practices during the last 10 years of normal operation of pressurised water reactors (PWRs) (Ref. 49). In addition, during Step 4, the TSC reviewed IAEA's International Reporting System (IRS) for events involving radioactive waste and spent fuel, and radiological protection aspects of these events (Ref. 55).

37 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. 50).
- NT / TUV SUD undertook a detailed technical review of general radiological protection, and in particular, optimisation of high dose work activities and impacts of accidents to people on-site (Ref. 51).
- GRS undertook a detailed technical review of criticality control in the spent fuel pool (Ref. 53), and ND's assessment of this topic is reported in Appendix A.
- REACT Engineering undertook a detailed technical review of decontamination and decommissioning (Ref. 54). The radiological protection aspects of these topic areas are summarised in my report. However, most of the TSC's work is reported in the Step 4 Radioactive Waste and Decommissioning Assessment Report of the UK EPR (Ref. 56).
- 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. 46.

38 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 UK EPR 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. 46.

39 The visibility of TSC work and feedback on progress and outcomes of TSC work was provided to EDF and AREVA throughout the process.

2.3.4 Cross-cutting Topics

40 The following Cross-cutting Topics have been considered within this report.

- Boron Dilution (see Section 4.11.1.1 and Appendix A).
- Source Terms (see Section 4.1).

2.3.5 Integration with other Assessment Topics

41 Radiological protection interfaces with all the other assessment topics, although there was only minimal interface with electrical power supply systems, management of safety and quality assurance, and security. The interfaces between 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 UK EPR (Ref. 1). In addition, each of the additional areas for Step 4 Radiological Protection assessment interfaced with Phase 2 (site licensing).

42 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.
- PRMS: 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.
- Optimisation for jumpers (maintenance activities involving jumpers during mid-loop operations with regard to fitting of SG nozzle dams are outside the scope of GDA): mechanical engineering and human factors.
- 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.

43 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 report for the UK EPR (Ref. 46): internal hazards, external hazards, PSA, fault studies, reactor chemistry and environmental issues.

44 As part of interfacing with other assessment topics, an RO on source terms (RO-UKEPR-73 (Ref. 9), see Section 4.1.1.1) was raised jointly with reactor chemistry and radioactive waste and decommissioning assessors plus assessors from the Environment Agency. Another RO on decontamination (RO-UKEPR-77 (Ref. 9), see Section 4.7.1) was raised

jointly with radioactive waste and decommissioning assessors plus assessors from the Environment Agency.

45 In view of the interfaces 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 (mainly associated with PWRs) operated by a range of companies across Europe (Great Britain, France, Germany, Sweden) and the United States. Assessors from the Environment Agency joined ND on most of the site visits. The purpose of these site visits was to benchmark the design, layout and operation of nuclear radioactive waste facilities associated with PWRs to assist us in the assessment of such facilities during GDA.

46 The benchmarking site visits identified examples of relevant good practice with regard to radiological protection and / or radioactive waste and decommissioning associated with PWRs. The full list of examples of relevant good practice is in Table 3 of the Step 4 radioactive waste and decommissioning assessment report (Ref. 56). 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 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 dose rates.
- The mapping of the radiological condition of the plant can take significant resource.
- The minimisation of Stellite™ reduces doses to workers and appears to be practical from an engineering point of view.
- With a suitably shielded design, access into containment at power 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 same 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.

47 Some of these examples of relevant good practice were helpful in my assessment of other topic areas as well as in the design, layout and operation of nuclear radioactive waste facilities.

2.3.6 Out of Scope Items

48 The following items have been agreed with EDF and AREVA as being outside the scope of GDA.

- Operational feedback data from the French fleet of NPPs provided during Step 4 is out of scope since future operators should not have to justify their operational practices against EDF operational practices (although some operational feedback data has been provided as examples of relevant good practice).
- Selection of operating equipment and comparison of existing suppliers are out of scope (for example, the PRMS and robots for work on SGs). (Access routes and locations for testing, maintaining and reading outputs from equipment are in scope.)
- Some details of operation and maintenance practices are out of scope (for example, although the UK EPR could accommodate mid-loop operations with regard to fitting of SG nozzle dams, it would be for future operators to justify such practices).
- Decontamination practices are out of scope (since these strongly depend on the nature of contamination and operator practices). (Decontamination facilities set aside for decontamination of equipment / plant are in scope.)
- Temporary shielding is out of scope (since this requires knowledge of hot spots or activities within, for example, neighbouring workshops). (Space for storage of temporary shielding is in scope.)
- Individual dose and its optimisation are out of scope (since future operator practices regarding dose sharing for tasks are not known). (Doses received by workers by virtue of them being at particular locations within the plant and subject to ambient dose rates are in scope.)
- Optimisation of dose in accidents is out of scope (as the precise location of workers is not known in advance). (The nature and location of evacuation routes to minimise doses to workers during accidents, and doses to operators required to bring the plant into a safe state, are in scope.)
- Protection of the public from planned radioactive discharges during normal operation is out of scope of ND's assessment (since this is in scope for the environmental assessment by the Environment Agency). (Direct shine from direct radiation originating from within the site during normal operation and accident conditions is in scope.)
- Site-specific Level 3 PSA is out of scope. (Generic Level 3 PSA is in scope and is covered in the Step 4 PSA assessment report on the UK EPR (Ref. 46).)

3 EDF AND AREVA'S SAFETY CASE

49 Radiological protection requirements were contained in Chapter 12 (entitled Radiation Protection) of the PCSR (Ref. 11). Sub-Chapters 12.0 to 12.5 dealt with radiation protection requirements, radiation protection approach, definition of radioactive sources in the primary circuit, radiation protection measures, dose uptake optimisation, and post-accident accessibility, respectively.

50 The PCSR (Ref. 11) was revised to take account of information / documentation requested and assessed by ND, and this cPCSR (Ref. 13) was submitted in March 2011 with the revised Master Submission List (Ref. 16).

3.1 Radiation Protection Requirements

51 In Sub-Chapter 12.0 (Ref. 11), Radiation Protection Requirements, EDF and AREVA outlined the relevant regulatory requirements in terms of international recommendations (Ref. 14), the European Directive which takes account of those international recommendations (Ref. 15), and the legislation that implements that Directive in England and Wales which is relevant to nuclear plant. This legislation was as follows.

- IRR99 (Ref. 17).
- REPPIR (Ref. 18).
- The Radioactive Substances (Basic Safety Standards) (England and Wales) Direction 2000 – this piece of legislation has since been subsumed by EPR10 (Ref. 19) and is enforced by the Environment Agency.

52 Sub-Chapter 12.0 (Ref. 11) also summarised definitions of zoning for areas to describe ranges of radiation dose rates, airborne contamination and surface contamination within the workplace.

53 Sub-Chapter 12.0 in the cPCSR (Ref. 13) was essentially unchanged (with minor editorial changes).

3.2 Radiation Protection Approach

54 In Sub-Chapter 12.1 (Ref. 11), Radiation Protection Approach, EDF and AREVA outlined the principles of radiation protection (justification, optimisation and limitation) and proposals to minimise radiation exposure and plant radioactivity levels by optimising the plant radioactivity inventory and shielding provisions. It stated that the individual dose limit for the plant would be 10mSv over 12 consecutive months (this was a company operational dose limit, not a legal dose limit), and the collective dose target would be 0.35person-Svy^{-1} per unit (i.e. per reactor). The collective dose target was based on current operational feedback from French NPPs. It was averaged over 10 years to take account of two refuelling only outages (ROO), three normal refuelling outages (NRO) (involving partial inspection) and one 10-year in-service inspection outage (ISIO) during the 10-year cycle. The main measures to optimise doses and to comply with this limit and target were summarised by EDF and AREVA as follows.

- Taking account of feedback and best practice from the best operational NPPs.
- Reducing equipment maintenance requirements by choosing reliable and suitable materials (taking into account conventional safety and human factors).

- Choosing materials to reduce activated corrosion products (mainly cobalt isotopes).
- Taking advantage of improvements in fuel assembly technology leading to fewer cladding defects and reduced risk of fission products entering the primary circuit.

55 Sub-Chapter 12.1 in the cPCSR (Ref. 13) was essentially unchanged (with minor editorial changes and updates to references).

3.3 Definition of Radioactive Sources in the Primary Circuit

56 In Sub-Chapter 12.2 (Ref. 11), Definition of Radioactive Sources in the Primary Circuit, EDF and AREVA explained that realistic source term data (from corrosion product and fission product activities) were used to estimate occupational doses, whereas more conservative data were used to define the biological protection design source term (used as a design parameter for buildings, systems and shielding provisions). This Sub-Chapter summarised the source of the radionuclides in the primary circuit, and identified 33 radionuclides whose specific concentrations affected radiological protection under normal operation.

57 Sub-Chapter 12.2 in the cPCSR (Ref. 13) was essentially unchanged (with minor editorial changes, clarification of some text, inclusion of additional references, and deletion of text regarding the realistic source term that had been included in a different Sub-Chapter).

3.4 Radiation Protection Measures

58 In Sub-Chapter 12.3 (Ref. 11), Radiation Protection Measures, EDF and AREVA provided greater detail on zoning for radiation than had been provided in Sub-Chapter 12.0, including zoning required by IRR99 (Ref. 17) in terms of controlled areas and supervised areas. It also covered shielding provisions, ventilation, and provision for monitoring rooms, monitoring employees, and monitoring the unit through the PRMS.

59 Sub-Chapter 12.3 of the cPCSR (Ref. 13) contained a number of changes. In addition to minor editorial changes, corrections and clarifications of text, changes to references, and minor modifications / updates to tables and figures, the main changes included the following.

- Additional information on contaminated systems present in buildings and correction of main building structures.
- Clarification of the design of the ventilation systems.
- Additional information on the monitoring of large objects.

60 Although there were a number of changes to Sub-Chapter 12.3 of the cPCSR (Ref. 13), as far as I am aware, there were no safety significant changes.

3.5 Dose Uptake Optimisation

61 In Sub-Chapter 12.4 (Ref. 11), Dose Uptake Optimisation, EDF and AREVA outlined the method for detailed dose prediction analysis. The key factors were as follows.

- Collecting dose uptake statistics from the best performing 1300 MWe NPPs and 1450 MWe NPPs (these last were omitted from the PCSR (Ref. 11), but were included in the cPCSR (Ref. 13)).

- Identifying activities with high dose uptake during normal operation and outages, and selecting them for detailed dose optimisation.
- Predicting dose uptake following detailed dose optimisation, taking into account the type of outage.
- Predicting the annual collective dose over a 10-year cycle.

62 This Sub-Chapter (Ref. 11) identified the reference dose (i.e. the collective dose from the best-performing French NPP), identified work activities that together were responsible for 50% of that collective dose, and optimised those doses to predict dose estimates. Those dose estimates were identified in three ways.

- Initial Predicted Dose Estimate (EDPI) that considered proven modifications.
- Optimised Predicted Dose Estimate (EDPOa) that considered proven modifications plus modifications being studied.
- Optimised Predicted Dose Estimate (EDPOc) that considered proven modifications, modifications being studied plus modifications still to be studied.

63 The work activities that together were responsible for 50% of the collective dose were identified by EDF and AREVA as follows (Ref. 11).

- Thermal insulation operations.
- Worksite logistics.
- Valve activities.
- Steam generator worksite.
- Worksite for opening and closing the reactor vessel.
- Fuel posting out worksite.
- Waste treatment operations.

64 The results of the EDF and AREVA's optimisation study (Ref. 11) were that the reference dose was 0.440 person-Svy^{-1} per unit. The estimated doses were predicted as follows: EDPI was 0.361 person-Svy^{-1} per unit, EDPOa was 0.345 person-Svy^{-1} per unit, and EDPOc was 0.331 person-Svy^{-1} per unit. These results were consistent with the collective dose target of 0.35 person-Svy^{-1} per unit that was stated in Sub-Chapter 12.1 (Ref. 11). These doses were averaged over 10 years to take account of the three types of outage already described.

65 Sub-Chapter 12.4 of the cPCSR (Ref. 13) contained a number of changes. In particular, dose optimisation studies had been completed which led to revisions of dose estimates. In addition to minor editorial changes and inclusion / update of references, the main changes included the following.

- Additional information on dose reductions associated with the source term following completion of dose optimisation studies.
- Clarification of the access to the Reactor Building during operation, and additional information on dose uptake linked with activities in the Reactor Building during operation.
- Update of the EDPO definition. The PCSR (Ref. 11) referred to the EDPI and two sub-types of the EDPO, namely EDPOa and EDPOc. Since the dose optimisation

studies had been completed, EDPOa and EDPOc had been consolidated into a single EDPO (Ref. 13).

- Update of information that took account of validated modifications following completion of the dose optimisation studies.
- The sections of Sub-Chapter 12.4 that covered the seven work activities that together were responsible for 50% of the collective dose were largely re-written to include the findings from those dose optimisation studies.
- The updated EDPI and EDPO in the cPCSR (Ref. 13) following completion of the dose optimisation studies were as follows. The reference dose was unchanged at 0.440 person-Svy^{-1} per unit. The final updated EDPI was little changed at 0.363 person-Svy^{-1} per unit. The final updated EDPO was little changed at 0.340 person-Svy^{-1} per unit, which was consistent with the collective dose target of 0.35 person-Svy^{-1} per unit. These doses were averaged over 10 years to take account of the three types of outage already described.

66 In summary, although there were a number of changes to Sub-Chapter 12.4 of the cPCSR (Ref. 13), as far as I am aware, there were no safety significant changes.

3.6 Post Accident Accessibility

67 In Sub-Chapter 12.5 (Ref. 11), Post Accident Accessibility, EDF and AREVA defined the systems, rooms and components for which access would be required in post-accident conditions for long-term cooling of the plant and fuel pond (over a period of one year), and identified work that would be necessary before access could take place. It also identified relevant legislation (IRR99, (Ref. 17); REPPIR, (Ref. 18); NIA65 (Ref. 20), and upper dose levels for personnel involved in intervention and in saving life.

68 Sub-Chapter 12.5 of the cPCSR (Ref. 13) contained a number of changes. In particular, some sections were revised in light of the findings of the study on post-accident accessibility performed for Flamanville 3 European Pressurised-water Reactor (FA3 EPR). In addition to minor editorial changes, corrections and clarification of text (including revision of some section headings), and inclusion of references, the main changes included the following.

- Additional information regarding access to the Safeguards Buildings and the Fuel Building in a long term post-accident situation.
- Additional information regarding dose management during accident conditions (relating to provisions under REPPIR (Ref. 18)).
- Additional information on access required to repair / maintain plant / equipment after one year following an accident, and measures taken to ensure that access is possible in respect of dose management.
- Clarification of the types of accidents during which intervention may be required.
- Additional information from the study performed for post-accident accessibility for the FA3 EPR.

69 In summary, although there were a number of changes to Sub-Chapter 12.4 of the cPCSR (Ref. 13), as far as I am aware, there were no safety significant changes.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR RADIOLOGICAL PROTECTION

70 The Step 4 Radiological Protection Assessment Plan (Ref. 1) 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.

71 At the beginning of the assessment, I prioritised four topics for assessment by TSCs. The first topic was shielding since this provided a key means of restricting exposure of workers and members of the public to direct radiation sources on the plant. It also had the advantage of being a discreet topic suitable for in-depth technical specialised analysis. The second topic was high dose work activities since these had a great deal of scope for reducing and minimising radiation exposure, and the opportunity was taken to involve specialist support with expertise in dose management at PWRs. The third topic was doses to people on-site during potential accidents since much could be achieved to minimise such doses through design. The fourth topic was criticality since this was an important factor in fresh and spent fuel storage, and also had the advantage of being a discreet topic suitable for in-depth technical specialised analysis.

72 The gathering of information to enable the TSCs to begin their assessment work was achieved through a TQ being raised for each topic. These TQs were broad in content to provide an overview of information and a framework from which to undertake the specialist assessment.

73 One of the key concerns identified in the Step 3 Radiological Protection Assessment Report (Ref. 6) was dose management of jumpers carrying out tasks at the SG worksite, and so dose optimisation for jumpers for this and other such tasks was afforded a specific topic description. Towards the end of the Step 4 assessment, EDF and AREVA declared that operation and maintenance practices involving fitting of steam generator nozzle dams were out of scope. The reason was that although the UK EPR could accommodate mid-loop operations with regard to fitting of SG nozzle dams, these were not essential for maintenance and it would be for future operators to justify such practices, and therefore these practices were outside the scope of GDA. In view of this, the two topics covering optimisation for work activities and optimisation for jumpers identified in the Step 4 Radiological Protection Assessment Plan (Ref. 1), and outlined in Section 2.3.2 above, are merged and reported as one topic in my assessment summary below.

74 My assessment of radiological protection has been supplemented by the assessment performed by the TSCs. During my assessment I raised a number of TQs and two ROs. Both of the ROs and some of the TQs were raised in collaboration with assessors in other topic areas. All 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). A summary of my assessment and its findings are presented below, with the exception of the assessment of criticality control in the spent fuel pool which is reported separately in Appendix A.

4.1 Normal Operation – Radiation Sources

4.1.1 Assessment – Radiation Sources

75 My assessment considered radiation sources from two perspectives. The first was to assess the management of the source term information, and the second was to assess reductions in the source term through selection of materials associated with the primary coolant.

4.1.1.1 Assessment - Information on the Source Term

76 The definition and appropriate use of the source term is an important stage in understanding and deriving the safety requirements of any nuclear activity. In the PCSR (Ref. 11), source terms are radioactive inventories which 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 more realistic inventories are required.

77 ND and the Environment Agency recognised that there was some consistency between the source terms used in different assessment areas, but there were also some apparent inconsistencies, and it was not always obvious how consistency was intended to be maintained. 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 to ensure consistently across the assessment areas.
- How the source term had been manipulated for use in each specific assessment area along with assumptions used.

78 The management of source term information is a cross-cutting issue that affects several assessment areas, and so ND and the Environment Agency raised RO-UKEPR-73 (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.

79 The RO stated that ND and the Environment Agency did not consider that EDF and AREVA 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), ROA-UKEPR-73-A.1 (Ref. 9), required EDF and AREVA 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). In addition to this assessment report, this aspect of source terms was discussed in ND's assessment reports on reactor chemistry (Ref. 79), radioactive waste and decommissioning (Ref. 56) and mechanical engineering (Ref. 100).

80 EDF and AREVA provided the response to this RO through three letters. The first letter, ND(NII)EPR00518N (Ref. 57), outlined the response plan. The second letter, ND(NII)EPR00554N (Ref. 58), provided a description of source term management and organisation in EDF and AREVA, explained the primary source term definition, and

provided references on the primary source term in the UK EPR (Ref. 59), the primary nuclide source term derivation within systems (Ref. 60), specific activity concentration of nuclides in reactor building systems (Ref. 61), and activity concentrations in a range of systems (Ref. 62). The third letter, ND(NII)EPR00627R (Ref. 63), provided a reference on the use of the source term in different assessment areas (Ref. 64).

4.1.1.2 Expectations – Information on the Source Term

81 The expectations of ND and the Environment Agency regarding information on the source term were identified in ROA RO-UKEPR-73.A1 (Ref. 9), and we expected EDF and AREVA to demonstrate the following points.

- How the source term has been derived (i.e. first principles, comparison with other reactors).
- Identify the assessment where the source term is used (e.g. waste management, discharges, normal operations, accident conditions).
- How the source term has been used consistently across the assessment areas.
- How the source term has been used in each specific assessment area.

4.1.1.3 Findings – Information on the Source Term

82 ND and the Environment Agency considered the information supplied in response to RO-UKEPR-73 (Ref. 9) and agreed that the evidence presented satisfied the regulatory expectations regarding derivation of the source term, identification of assessments where the source term was used, use of the source term consistently across assessment areas, and use of the source term in specific assessment areas.

4.1.1.4 Assessment - Reductions in the Source Term

83 During maintenance and repair work activities, worker exposure to radiation is mainly due to activated corrosion product deposits within the primary circuit of the PWR which make a major contribution to dose rates in the vicinity of systems and components. The reduction of contamination is therefore of prime importance. The selection of materials which result in lower levels of corrosion products capable of activation in the primary circuit, therefore, help to reduce dose rates in the vicinity of systems and components and thereby reduce worker radiation exposure.

84 Assessment of reductions in source terms and radiation doses arising from reductions in the use of cobalt, silver and antimony were outlined in my Step 3 Radiological Protection Assessment Report (Ref. 6). Evidence to demonstrate source term reduction through selection of materials associated with the primary coolant was considered during Step 4 (Ref. 1). I was supported in my assessment on radiation sources by ND's TSC, TUV SUD (Ref. 51). AMEC's literature review of radiological protection and radioactive waste and decommissioning practices during the last 10 years of normal operation of PWRs (Ref. 49) provided useful benchmark information. In addition, this topic was of direct interest to other GDA assessment areas, namely reactor chemistry, radioactive waste and decommissioning, and mechanical engineering.

85 In addition to Sub-Chapter 12.1 on radiation protection approach and Sub-Chapter 12.3 on radiation protection measures of the PCSR (Ref. 11), EDF and AREVA also provided documentation to demonstrate compliance with guidance expectations in RP.1 of the

SAPs (Ref. 4). These covered a methodical note to define iodine risk and / or aerosol risk rooms in FA3 (Ref. 72), radiation protection guidelines (Ref. 67), collection of requirements and elements of methodology for radiation protection zoning in FA3 (Ref. 69), and reactor building specifications (Ref. 73).

4.1.1.4.1 Cobalt Reduction

86 In many nuclear power plants (NPPs), activated corrosion products in the primary coolant increase dose rates through activation of cobalt-59 to cobalt-60 in the Stellite™ content of hard facings, and activation of nickel-58 to cobalt-58 in inconel 690 alloys and some stainless steels; cobalt-58 and cobalt-60 typically account for over 80% of equivalent dose rates associated with the primary coolant.

87 The cobalt content of construction materials in contact with primary coolant for UK EPR SGs was specified as follows (Ref. 11).

- Stainless steels or nickel-chromium-iron alloys (other than in tube bundle): cobalt content was up to 0.10 % (out of core); cobalt content was up to 0.06 % (in core).
- Tube bundle: cobalt content was up to 0.015 %.

88 In French NPPs, the global Stellite™ inventory was not changed for the 1300MWe plants, but a decrease in the out-of-core cobalt-60 contamination was observed related to a decrease of the cobalt content of the SG tubes (Ref. 65). This experience showed that a decrease in the cobalt content of the tubes significantly reduced the level of cobalt-60 contamination. Further experience was also gained from German NPPs. Management of radiation exposure in German NPPs is considered to be amongst the leaders worldwide following Stellite™ replacement and was considered ALARP (Ref. 93). Therefore, provided that the level of cobalt in the UK EPR was in the range of the levels of cobalt in the latest German PWRs, there was no reason to seek further reductions in the level of cobalt in the UK EPR. The experience from German NPPs showed that a progressive reduction of the Stellite™ content, in cases where the global surface in contact with primary coolant was below approximately $2m^2$, had little impact on dose rates since the dose rate was relatively insensitive to the amount of Stellite™ present (Ref. 65). The contact surface with the primary coolant for the UK EPR was estimated as approximately $1.9m^2$ (excluding control rod driving mechanisms), and so EDF and AREVA concluded that the doses in the vicinity of primary components should be in the same range as for the latest generation of German NPPs.

4.1.1.4.2 Silver and Antimony Reduction

89 In many NPPs, activated corrosion products in the primary coolant increase dose rates through activation of silver-109 to silver-110m in helicoflex seals, and activation of antimony-123 to antimony-124 in bearings, and these activated corrosion products contribute to worker exposure from the primary coolant.

90 To avoid such activation of the primary coolant in the UK EPR, silver and antimony were excluded as far as possible as constituents of materials in contact with the primary coolant or within the Reactor Building (Ref.65). Use of silver and antimony was only permitted in cases where there were no suitable substitute materials, and where their use was temporary and controlled. Since exclusion of silver and antimony from components in contact with the primary coolant is dependent on the procurement process, a list of

elements which may become activated will need to be provided in documentation sent to potential suppliers.

91 EDF and AREVA identified potential sources of silver causing contamination in existing NPPs as follows (Ref. 65).

- Silver may be released from neutron-absorbing alloys containing silver-indium-cadmium in control rods released via cladding defects before the cladding received a special treatment (nitride coated control rods).
- Silver may be released from reactor pressure vessel head seals if they are damaged during mounting / dismounting maintenance operations.
- Silver may be released from silver coated seals in auxiliary systems.

92 To restrict the release of silver, feedback from French NPPs has shown the following improvements.

- No perforation had been identified on nitrided coated control rods, such that they were no longer regarded as potential sources of contamination.
- In the UK EPR, the reactor pressure vessel seal was incorporated into the closure head so that it was in a restricted space and not in contact with the primary coolant. Also, the seal was changed at each shutdown. Moreover, all of the French NPPs were fitted with reactor pressure vessel seals coated with silver, and experience showed that the seals were not systematically contaminated.
- In the UK EPR, the seals in auxiliary systems had been replaced by graphite seals, and flange type design of piping had been replaced by welded type design of piping. Antimony-bearing alloys had also been avoided in the auxiliary system of the UK EPR (Ref. 65).

93 Therefore, the silver and antimony content of materials in contact with the primary coolant was reduced as far as was reasonably practicable, and the need for any further reductions in the levels of silver and antimony were not considered necessary.

4.1.1.5 Expectations - Reductions in the Source Term

94 My Step 4 Assessment Plan (Ref. 1) explained that my assessment of radiation sources would include the following matters.

- The reactor, nuclear fuel and radiation sources.
- Radionuclides in the primary circuit and fuel route.
- Activation of systems, structures and components.

95 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, and 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. 33), where it advises that consideration should be given to minimising the formation of activated corrosion products from circuit components.

4.1.1.6 Findings - Reductions in the Source Term

96 I considered the information regarding reductions in the levels of cobalt, silver and antimony from the source term in the UK EPR (Ref. 65), and concluded that the reductions incorporated in the design compared with previous plants appeared ALARP and therefore further reductions of these elements from materials associated with the primary coolant were not necessary. Nevertheless, the restriction of exposure through material selection is dependent on procurement procedures. In addition, new materials may be developed before a UK EPR is constructed, in which case it would be appropriate for a further review of materials to be undertaken before future procurement. I have captured this requirement in a GDA assessment finding, **AF-UKEPR-RP-01**, below (see Section 4.1.3).

4.1.2 Expectations – Summary for Radiation Sources

97 Expectations for information on the source term were identified in ROA RO-UKEPR-73.A1 (Ref. 9). Expectations for reductions in the source term were in para. 479 of ND's SAPs (Ref. 4) and in para. 4.1 of the TAG on radiological protection (Ref. 33).

4.1.3 Findings – Summary for Radiation Sources

98 In my opinion and in the opinion of the TSC, the evidence to substantiate the arguments relating to radiation sources regarding information on the source term, and reductions in the source term through selection of materials associated with the primary circuit, was suitable and sufficient.

99 I have identified a GDA assessment finding regarding materials selection associated with the primary coolant.

> ***AF-UKEPR-RP-01:*** *The licensee shall provide procurement procedures that require a review of materials associated with the primary coolant before purchase of those materials from their supplier in order to identify if there are any improvements in reductions in levels of cobalt or any other elements in materials which might lead to further reductions in radiation exposure of workers, and which would not compromise the functionality of those materials. This shall be complete before mechanical, electrical and control and instrumentation systems are delivered to site.*

4.2 Normal Operation – Designated Areas (Radiological Classification of Areas / Radiological Zoning)

4.2.1 Assessment – Designated Areas

100 Assessment of designated areas was outlined in my Step 3 Radiological Protection Assessment Report (Ref. 6). Evidence to demonstrate designated areas was considered during Step 4 (Ref. 1). I was supported in my assessment by ND's TSCs, NT (Ref. 50) and TUV SUD (Ref. 51).

101 The contamination zoning aspects of this topic is linked to Section 4.4 on contaminated areas in this assessment report.

4.2.2 Background - Radiological Classification of Areas

102 IRR99 require some areas to be designated as controlled or supervised areas (Ref. 17), 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 France.

103 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.

104 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.

105 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.

4.2.2.1 Assessment - Radiation Zoning Methodology

106 The Flamanville 3 (FA3) EPR was the reference design for the UK EPR, and so the radiation zoning methodology and the radiological zoning classification scheme for the FA3 EPR design demonstrated the likely methodology and scheme for the UK EPR.

107 EDF and AREVA's radiation protection guidelines (Ref. 67) and radiological zoning reports (Refs 68 and 69) outlined how the FA3 EPR was divided up into areas consistent with requirements in IRR99 (Ref. 17). These areas were as follows.

- Zones where radiation doses should not exceed 1mSv per year, and the effective dose rate should not exceed 0.08mSv per month.
- Monitored zones where radiation doses may be between 1mSv and 6mSv per year, and the effective dose rate should not exceed 7.5microSv per hour.
- Controlled areas within which radiation doses may exceed 6mSv per year, and the effective dose rate may exceed 7.5microSv per hour.

108 The nuclear island comprises the Reactor Building, Safeguard Buildings, Fuel Building, Nuclear Auxiliary Building, Access Tower Building, Diesel Buildings and Effluent Treatment Building. The controlled area within the nuclear island was subdivided into green, yellow, orange and red zones based on increasing dose rates (denoted as A, B, C, D, E and F) and access restrictions. The zones were as follow.

- Green: $A \leq 10 microSvh^{-1}$; $2.5A \leq 25 microSvh^{-1}$.
- Pale yellow: $B \leq 0.1 mSvh^{-1}$; $2B \leq 0.2 mSvh^{-1}$.
- Deep yellow: $C \leq 1 mSvh^{-1}$; $2C \leq 2\ mSvh^{-1}$.

- Orange: D ≤ 10mSvh^{-1}; 3D ≤ 30mSvh^{-1}; E ≤ 0.1Svh^{-1}.
- Red: 3E ≤ 0.3Svh^{-1}; F ≤ 1Svh^{-1}.

109 Design objectives outlined in Sub-Chapter 12.3 of the PCSR (Ref. 11), and in the document that presented requirements and elements of methodology for radiation protection zoning for the FA3 EPR (Ref. 69), stated that high occupancy areas, such as corridors, floors, stairwells, elevators and frequently used passageways were shielded to meet the dose rate criterion Green A (≤ 10microSvh^{-1}). The documentation also stated that secured corridors and emergency stairwells were shielded to meet the dose rate criterion Green 2.5A (≤ 25microSvh^{-1})

110 In addition, these zones could have contamination requirements superimposed upon them.

111 In addition to Sub-Chapter 12.3 of the PCSR on radiation protection measures (Ref. 11) and the requirements and elements of methodology for radiation protection zoning for the FA3 EPR (Ref. 69), EDF and AREVA also provided documentation to demonstrate compliance with guidance expectations in RP.3 of the SAPs (Ref. 4). These covered the individual and collective radiation protection measurement system (Ref. 70) and a cleanliness / waste design guide for the buildings of FA3 EPR (Ref. 71).

4.2.2.2 Assessment – Dose Rates for Occupied Areas and Access to Controlled Areas

112 I raised TQ-EPR-1229 (Ref. 8) on dose rates for occupied areas to request information / documentation that summarised dose rates and radiological classifications (zones) for all modes of plant operation within the Reactor Building, Fuel Building, Safeguards Building, Auxiliary Building and Waste Building. The TQ also requested that, as a minimum, the information for each room should include the following.

- Room description and number / designation.
- Radiological classification (namely dose rate criteria).
- Dose rate prediction(s) for each room giving the maximum dose rate present during all modes of operation (for example, power operation, outages, refuelling).
- Reference to shielding assessments / calculations containing data regarding the assumed radiation sources, shielding provisions and calculated dose rates.

113 EDF and AREVA were unable to provide this information / documentation for the FA3 EPR or UK EPR within the timescale of GDA because it took a period of time for ND and EDF and AREVA to come to a common understanding on the evidence required.

114 I also raised TQ-EPR-1453 (Ref. 8) on access to controlled areas to request information / documentation on the layout of areas set aside for accessing controlled areas, plus information / documentation showing possible locations for relevant health physics facilities for entering and exiting controlled areas. The areas set aside for accessing controlled areas and the location and layout of health physics facilities need to be sympathetic with the radiological zoning classification scheme of the plant (e.g. health physics provisions need to be suitable and sufficient to support entry into radiological zones with potentially high levels of surface and / or airborne contamination).

115 The response to TQ-EPR-1453 (Ref. 8) showed that access to the nuclear island controlled area will be through the Access Tower. Workers will have to present their badges, dosemeters and barcodes corresponding with their work activities in order to pass through the controlled area access modules. The locations of rooms for the

management of access control and for health physics work planning and supervision, and the potential location of equipment such as body contamination monitors, decontamination showers and stores for personal protective equipment (PPE) (including respiratory protective equipment (RPE)), were provided.

116 As I raised TQ-EPR-1453 (Ref. 8) towards the end of my assessment phase, there was insufficient time available for discussions with EDF and AREVA on the information that they had provided. From this information, it was not clear how these various rooms and pieces of equipment were linked together to form a logical path for entering and exiting the controlled area in order to minimise the spread of contamination. Also, there was little information on male and female changing and washing facilities, locations for issue of PPE / RPE, storage areas for personal issue PPE such as hard hats and safety boots, and the position of the contamination control (jump) barrier into the controlled area (there was an example of a sub-change area including a contamination control (jump) barrier into an area with greater potential contamination). However, the level of detail provided was not unexpected at this stage in the design since such matters are largely the responsibility of future licensees.

117 The key feature for the design is to provide sufficient space at suitable locations for the necessary rooms and pieces of equipment. Such facilities also need to be able to accommodate the additional workloads which may be put upon them by intervention personnel during accident conditions (see Section 4.12.3 below. As more information will be required to form a view on the adequacy of the health physics and changeroom facilities on a site-specific basis, I captured this requirement in a GDA assessment finding, **AF-UKEPR-RP-02**, below (see Section 4.2.4).

4.2.3 Expectations – Designated Areas

118 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.

119 RP.3 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.

120 Paras 4.6 and 4.7 of the TAG on Radiological Protection (Ref. 33) 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's main 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.

4.2.4 Findings – Designated Areas

121 Typically within the UK, the initial dose rate criteria for all rooms of a facility are outlined in a designation of areas (often referred to as a radiological classification of areas) document. Shielding assessments are then carried out to ensure that the shielding provisions reduce dose rates in the room to within the criteria outlined in the designation of areas. Where initial dose rate predictions do not meet the criteria, shielding provisions may be revised to further reduce dose rates to within the criteria. Alternatively in cases where changes to the shielding are not practicable, the radiation zoning classification may be increased to reflect the potential for high dose rates, along with further restrictions on occupancy applied to ensure dose accrual remains acceptable. Shielding summary documents are usually produced to confirm that conservative dose rate predictions acceptably meet the radiological zoning criteria for each room of the facility. This should also include reference to the detailed shielding assessment from which the results have been extracted.

122 Given the above methodology, the GDA assessment had initially intended to perform a high level review of shielding provisions and dose rate profile across the nuclear island for all modes of operation. I raised TQ-EPR-1229 (Ref. 8) to request documentation outlining the current radiological zoning for the UK EPR, along with a shielding summary showing that the theoretical dose rates calculated in shielding assessments met the radiological zoning criteria.

123 EDF and AREVA were unable to provide this information / documentation for the FA3 EPR or UK EPR within the timescale of Step 4 of the GDA assessment because it took a period of time for ND and EDF and AREVA to come to a common understanding on the evidence required. In the absence of a reference radiological zoning scheme and any shielding summary documentation for the UK EPR, my assessment was unable to determine whether the general shielding design and dose rate profile throughout the UK EPR were acceptable when compared to UK design guidance and practices.

124 Radiological zoning for restriction of exposure to ionising radiation of workers is fundamental to the basic design of the nuclear island of the UK EPR. In addition, bulk shielding is inextricably linked with civil engineering aspects of the UK EPR design, and bulk shielding assessments need to be completed before nuclear island construction commences. Therefore, in my opinion, suitable and sufficient detailed work should be completed within GDA to demonstrate that the bulk shielding provided by nuclear island construction concrete is adequate.

125 I identified the following GDA Issue (**GI-UKEPR-RP-01**) and GDA Issue Action (**GI-UKEPR-RP-01.A1**) regarding radiological zoning and bulk shielding. The complete GDA Issue and associated actions are formally defined in Annex 2 of this report.

> ***GI-UKEPR-RP-01:*** *Radiological zoning for restriction of exposure to ionising radiation of workers is fundamental to the design of the nuclear island, and bulk shielding is inextricably linked with civil engineering aspects of that design. The radiological zoning classification scheme underpinned by design shielding calculations is not referenced in the GDA submission for the UK EPR design.*
>
> ***GI-UKEPR-RP-01.A1:*** *Provide an overview document that supplements the claims and arguments presented in the PCSR Chapter 12.3 with additional information on the radiological zoning classification scheme for the nuclear island, including dose rate criteria and predictions for all modes of plant operation, for occupied areas as a direct reference from the PCSR.*

126 The GDA Issue and GDA Issue Action on radiological zoning and bulk shielding covered the areas requested in TQ_EPR_1229 above (with the exception that I do not consider it necessary to assess the Waste Building during GDA since this would be largely site-specific).

127 In my opinion, form the evidence provided, it was not clear how rooms and equipment were linked together to form a logical path for entering and exiting the controlled area in order to minimise the spread of contamination (see Section 4.2.2.2 above). In light of the responsibility of future licensees for health physics facilities and arrangements to ensure that they are suitable and sufficient for controlling exposures of workers and for minimising the spread of contamination, I have identified a GDA assessment finding on access to controlled areas and health physics facilities as follows.

> ***AF-UKEPR-RP-02:*** *The licensee shall provide a report to demonstrate that the layout of the health physics facilities for entering and exiting the controlled area are suitable and sufficient, in particular, provision of rooms and equipment (including PPE / RPE) shall be arranged to minimise the potential spread of contamination. The report shall also include the ability of the facilities to accommodate additional workloads which may be put upon them by intervention personnel during accident conditions. This shall be complete before first structural concrete.*

4.3 Normal Operation – Shielding

128 The shielding assessment was undertaken to assess the UK EPR shielding provisions in support of the PCSR submission (Ref. 11) to review the arguments presented in the PCSR and to assess whether the evidence presented substantiated those arguments for shielding. The objectives of the shielding assessment were as follows.

- To be satisfied that the UK EPR shielding design fulfilled the requirements outlined in the SAPs (Ref. 4), in particular RP.6, and in the TAG for radiation shielding (Ref. 29).
- 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 Assessment – Shielding

129 This part of my assessment outlined the design criteria used to assess the acceptability of shielding provisions. 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. 74), the SAPs (Ref. 4) and the TAG on radiation shielding (Ref. 29).

130 To begin my assessment, I raised TQ-EPR-594 (Ref. 8) that requested 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 matters as indicated below which were taken from my Step 4 Assessment Plan (Ref. 1).

- Shielding to protect workers and the public from direct radiation during normal operation.
- Shielding during maintenance and to allow access into the Reactor Building at power.
- Shielding for enclosures (e.g. cells, glove boxes).

- Shielding materials and neutron activation of shielding.
- Shielding calculation methods and computer codes.

131 The topic of shielding was assessed by ND's TSC, NT (Ref. 50). The information and references provided in response to TQ-EPR-594 (Ref. 8) were used as the groundwork for the assessment. The general areas considered within the shielding assessment were as follows.

- Design Criteria.
- Shielding Design Basis Data (Shielding Source Terms, Shielding Materials).
- Radiological Classification of Areas (already discussed above in Section 4.2 of this assessment report).
- Calculation Methods (Computational Codes, Application in Shielding Assessments).
- Shielding Provisions (Bulk Shielding Provisions, Local Shielding and Penetration Assessments, Temporary Shielding Provisions).
- Dose Uptake.

4.3.1.1 Design Criteria

132 The design criteria used by the TSC for its assessment were dose limits and ALARP requirements in IRR99, and dose targets during normal operation, namely, NT.1 Target 1, NT.1 Target 2 and NT.1 Target 3 on any person on the site, any group on the site and any person off the site, respectively. These criteria are discussed in detail in Section 2.2, Standards and Criteria.

4.3.1.2 Assessment - Shielding Design Basis Data

4.3.1.2.1 Assessment – Shielding Source Terms

133 Section 4.1 above of this assessment report considered radiation sources from two perspectives. The first was to assess the management of the source term information, and the second was to assess reductions in the source term through selection of materials associated with the primary coolant. In this part of my assessment report on shielding design basis data, a third perspective was considered, namely to view the source terms and physical data used as the basis for all shielding analyses, and to consider the nature and magnitude of the source terms used within the Reactor Building when operating at full power and during shutdown.

134 Shielding design is dependent on accurate yet suitable conservative radiation source terms. The shielding source terms were reviewed to establish that the most significant sources of radiation with respect to reactor design (as outlined in Ref. 74) were taken into account and were sufficiently conservative for shielding calculations and dose rate predictions, as advised in para. 4.1 of the radiation shielding TAG (Ref. 29).

135 Sub-Chapters 12.2 and 12.3 of the PCSR (Ref. 11) summarised the source terms associated with external dose accrual within the UK EPR during full power operation and shutdown. These source terms were generally presented as realistic source terms (based on feedback on mean activity levels from French and German NPPs) and biological shielding design source terms (based on maximum activity levels from French NPPs). The TSC's assessment considered information on source terms presented in documentation regarding radiation protection guidelines (Ref. 67) and primary source

term of the UK EPR reactor (Ref. 59) (as referenced in Section 4.1 on Radiation Sources in this assessment report). Other documentation covered modelling data and input data for calculating dose rates (Ref. 75) and responses to letters regarding input data concerning neutron sources for radiation protection instruction by the French Nuclear Safety Authority and its TSC (ASN and IRSN, respectively) (Ref. 76).

4.3.1.2.1.1 Reactor Building at Full Power Operation

136 When the reactor is at full power, the dose rates and shielding requirements within the Reactor Building near the primary circuit are driven by nitrogen-16 (gamma emitter) and nitrogen -17 (neutron emitter) in the primary coolant, and by the reactor core (gamma and neutron emission).

137 Sub-Chapter 12.3 of the PCSR (Ref. 11) summarised the methods used to calculate quantities of activation products in the primary coolant (nitrogen-16, nitrogen-17, tritium, argon-41 and carbon-14). The TSC considered that the evidence presented showed that the calculation methods for quantities of activation products was adequate (see para. 4.1.9 of the radiation shielding TAG, Ref. 29).

138 Calculation of neutron sources terms around the reactor pit used the shielding code Monte Carlo N-Particle (MCNP). Provided that the reactor MCNP calculations took adequate account of source terms and shielding provisions in and around the reactor pit, the neutron flux calculated at the boundary of the reactor pit can be used as the source term for subsequent neutron shielding calculations. I raised TQ-EPR-1230 (Ref. 8) on the reactor vessel boundary source term generation and TQ-EPR-1454 (Ref. 8) to clarify the information requested in the former TQ.

139 The responses to TQ-EPR-1230 and TQ-EPR-1454 were not received in time to consider them before the issue of the TSC's report (Ref. 50). Although details of these calculations were not available for review by the TSC, the TSC noted that the results of MCNP core calculations were provided in Ref. 75 which were then used to assess potential leak paths such as radiation streaming through primary circuit penetrations around the reactor pit (Ref. 59). The TSC also noted that secondary gamma radiation arising from neutrons interacting with matter such as shielding materials had been adequately taken into account when performing calculations using MCNP, as advised in para. 4.1.7 of the radiation shielding TAG (Ref. 29). Therefore, the TSC was sufficiently satisfied with the evidence provided to make supplementing the TSC's assessment with the responses TQ-EPR-1230 and TQ-EPR-1454 unnecessary. I concur with the opinion of the TSC, and the breadth of the TSC's assessment was consistent with ND's sampling approach to assessment.

4.3.1.2.1.2 Reactor Building During Shutdown

140 When the reactor is shutdown, the external dose rates within the Reactor Building are dominated by gamma radiation emitters in the primary circuit from activated corrosion products (e.g. cobalt-58 and cobalt-60) and fission produces as a result of fuel defects (see para. 4.1.8 of the radiation shielding TAG, Ref. 29). In contrast to this, neutrons are missing from the core during shutdown, and levels of activation products (e.g. nitrogen-17) will be negligible given the short half life of the most significant activation products.

141 Source terms for shielding design with sufficient consideration given to deposition (plate-out) of activity on the internal walls at various points within the primary circuit were also

presented by EDF and AREVA (Ref. 59), as advised in the radiation shielding TAG (Ref. 29).

4.3.1.2.1.3 Other Buildings and Other Sources

142 The source terms used for the other buildings of the nuclear island were based solely on gamma emitting radionuclides since these will dominate dose rates, as was the source term for the Reactor Building.

143 Evidence for spent fuel source terms was presented in the response to TQ-EPR-1352 (Ref. 8). This response stated that the spent fuel source term only considered contributions from fission products and heavy elements. Any additional contributions from activated elements or activated products were ignored based on feedback from French NPPs. The spent fuel source terms were calculated using the calculation codes APOLLO 2.5 and DARWIN 2.1.1, although the qualification reports for these codes were not provided for review. This aside, in the opinion of the TSC, the resulting source terms should be adequately conservative for use in shielding and dose rate calculations.

144 Evidence for derivation of activation source terms was presented in the response to TQ-EPR-1228 (Ref. 8). During outages, workers are likely to work in the reactor pool close to the reactor pressure vessel, therefore, activation of components were calculated to estimate dose rates and to design shielding. The activation of components was calculated using TRIPOLI models and DARWIN 2.2 calculations. This activation would be limited by the thick concrete walls of the reactor pit, resulting in activation of equipment or concrete structures behind these walls being negligible compared to other sources of radiation. Based on the information provided, the TSC considered that there was no reason why activation source terms would not be negligible.

4.3.1.2.2 Expectations – Shielding Source Terms

145 Expectations for source terms for shielding are covered in RP.6, Shielding, and para. 493 of the SAPs (Ref. 4), and in Section 4.1, Source Term Generation, of the TAG on radiation shielding (Ref. 29).

4.3.1.2.3 Findings – Shielding Source Terms

146 In the opinion of the TSC, no concerns were raised during the review of the source terms used in defining the shielding provisions for the UK EPR. It was apparent that the shielding source terms were upper bound and sufficiently conservative compared to realistic source terms derived from existing plant data. This gave confidence that the shielding provisions and predicted dose rates for any given area of the plant will also be conservative with respect to radiological protection. I concur with the opinion of the TSC from the evidence provided on shielding design basis data.

4.3.1.2.4 Assessment – Shielding Materials

147 In order to perform shielding calculations it is necessary to have a detailed knowledge of the compositions and densities of specific shielding materials since these have a significant effect on materials' shielding performance. Typical materials employed for shielding in the UK EPR are described in Sub-Chapter 12.3 of the PCSR (Ref. 11) as

standard concrete, steel, lead, shielded glass and hydrogenated neutron shielding materials.

4.3.1.2.4.1 Concrete Bulk Shielding

148 The majority of bulk shielding of the UK EPR is provided by the concrete walls and floor slabs of the civil structure which provided both gamma and neutron shielding. The neutron shielding performance of concrete is largely driven by the hydrogen content of the concrete, which can reduce as the concrete "dries out". Similarly, the gamma shielding performance of concrete slightly decreases as the concrete "dries out" due to the decrease in density.

149 The concrete composition of the UK EPR was summarised in Ref. 75. This showed that the assumed concrete composition of the UK EPR was conservative when compared to concrete compositions assumed in UK facilities since a lower hydrogen content by mass of concrete was assumed for the UK EPR. This meant that the neutron dose rates calculated using the UK EPR concrete composition would remain conservative over the full lifetime of a plant, even towards the end of a plant's operating life, by which time the concrete would be very dry with a concomitant reduction in neutron shielding performance.

4.3.1.2.4.2 Neutron Shielding Provision

150 In addition to concrete shielding, Ref. 9 showed that hydrogenous materials (e.g. polyethylene) and / or boronated materials were generally used for local neutron shielding provisions on items such as doors and gates.

151 Consideration was also given to the use of novel shielding materials where additional physical parameters, such as increased structural strength, or greater heat or chemical resistance, were required. For example, compositions of polymeric resins, such as boronated resin containing aluminium oxide, would provide shielding performance for both neutron and gamma radiation as well as greater heat resistance when compared with more conventional shielding materials (e.g. polyethylene). These examples were described in Ref. 59 and in the response to TQ-EPR-1228 (Ref. 8), and would be suitable for high temperature environments, such as around reactor pit penetrations.

4.3.1.2.4.3 Gamma Shielding

152 Materials for gamma shielding were identified in Sub-Chapter 12.3 of the PCSR (Ref. 11) and in the response to TQ-EPR-1228 (Ref. 8). The densities specified for steel, lead and anti-radiation glass (e.g. lead glass with density approximately $4gcm^3$) were the same as those used in UK shielding assessments. In addition, low-cobalt steels were used in locations where high neutron fluxes would be found to minimise the potential for increased gamma dose rates from activation of cobalt in the steel.

4.3.1.2.4.4 Liquid Shielding

153 Appropriate account was taken for use of water shielding in areas such as the spent fuel pool, and for the variation of water density within various sections of the primary coolant circuit. The use of liquid shielding materials (pools) was also used in radiological studies for handling equipment during outages (Ref. 77).

4.3.1.2.5 Flux to Dose Conversion Factors

154 Many radiation transport codes calculate particle flux, which is then converted to dose using energy dependent conversion factors. The response to TQ-EPR-1353 (Ref. 8) provided evidence that the flux to dose conversion factors were used in the UK EPR shielding assessment to determine neutron and gamma dose equivalent rates, which were taken from international advice from the International Commission on Radiological Protection (Ref. 78).

4.3.1.2.6 Expectations – Shielding Materials

155 Advice on shielding materials is provided in the TAG on radiation shielding (Ref. 29). Advice on solid, liquid and novel shielding materials is provided in paras 4.5, 4.6 and 4.7 of the TAG, respectively. Additional guidance on liquid shielding is provided in para. 495 of the SAPs (Ref. 4). Advice on flux to dose conversion factors is provided in para. 4.4.9 of the TAG. Advice on shielding being effective under all conditions is in RP.6 of the SAPs (Ref. 4).

4.3.1.2.7 Findings – Shielding Materials

156 In the opinion of the TSC, the composition and densities of shielding materials used in the UK EPR shielding calculations were adequately conservative and consistent with those typically used within the UK. Shielding materials were also adequately specified according to their radiation shielding performance and ability to maintain integrity in adverse conditions (e.g. high temperature environments). I concur with the opinion of the TSC from the evidence provided on shielding materials.

157 The response to the GDA issue, **GI-UKEPR-RP-01**, on radiological zoning and bulk shielding (see Section 4.2.4 above), will also provide further confirmatory information regarding the adequacy of materials used in bulk shielding.

4.3.1.3 Assessment - Radiological Classification of Areas

158 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.

159 This topic was discussed above in Section 4.2, Designated Areas, of this assessment report, and is included here for completeness to show that the radiological classification of areas fits into the overall shielding assessment.

4.3.1.4 Assessment - Calculation Methods

160 This part of my assessment reviewed the calculation methods and computational codes, and their adequacy for use in shielding assessments.

161 My assessment report summarises the computational codes used and their application in shielding assessments for the UK EPR.

4.3.1.4.1 Computational Codes

162 A number of computational codes were used in the shielding assessments for the UK EPR, and the majority of the neutron and gamma shielding calculations were undertaken using codes such as MCNP and PANTHERE. Advice on hand calculation methods and computer codes is provided in para. 4.4.1 of the TAG on radiation shielding (Ref. 29). The codes used are discussed below.

163 MCNP was used in shielding calculations involving neutron and secondary gamma radiation transport within the Reactors Building. This code has wide use within the nuclear industry over many years and has been subject to numerous validations studies and comparisons with benchmarking calculations. EDF and AREVA provided evidence regarding use of the MCNP code for the UK EPR in Sub-Chapter 12.3 of the PCSR (Ref. 11), the response to TQ-EPR-1227 (Ref. 8) and supporting documentation (Ref. 75). The TSC advised that the input data for MCNP calculations (e.g. geometry modelling, source terms, material cross-section data) were adequately defined for shielding design purposes.

164 PANTHERE was used to perform detailed calculations involving multiple gamma radiation sources and complex shield geometries within the UK EPR. This code is not commonly used in the UK, so the TSC undertook a more detailed review of the documentation supporting the use of the PANTHERE shielding code. The evidence regarding use of the PANTHERE code was provided in the responses to TQ-EPR-1226 (Ref. 8), TQ-EPR-1227 (Ref. 8) and other supporting documentation (Refs 67, 80, 81, and 82). The TSC initially had some concerns regarding radiation streaming involving multiple scatters through weaknesses in shielding, such as in thin gaps around doors. However, EDF and AREVA provided evidence showing that such gaps had been designed out by using sufficient shielding overlap, and this was confirmed by hand calculations performed by the TSC.

165 MicroShield® was used for gamma shielding calculations with simple source-shield geometries. This code has been widely used by the nuclear industry for a number of years and yields reliable results. The TSC confirmed that MicroShield® was an appropriate tool for simple bulk gamma shielding assessments in reactor design.

166 TQ-EPR-1227 (Ref. 8) and TQ-EPR-1228 (Ref. 8) requested information / documentation on computational codes and methods used for shielding calculations. The response to the TQ included a brief description of the TRIPOLI and DARWIN codes but no supporting references were supplied. The response to the TQ indicated that the TRIPOLI code was mainly used for particle transportation, and was a French standard equivalent to the USA Los Alamos MCNP code. The DARWIN code was used to undertake activation calculations for the UK EPR. The TSC was not familiar with either of these codes, and since no supporting references were submitted, the TSC was not able to review these codes. Nevertheless, the TSC had no reason to believe that these codes were not used appropriately.

4.3.1.4.2 Application of Codes in Shielding Assessments

167 Computational codes used in shielding applications only provide acceptable dose rate estimates when applied correctly to shielding assessments using appropriately conservative data and modelling assumptions. The detailed examination of code inputs for computational codes is not appropriate for assessment during GDA since EDF and AREVA's internal quality assurance procedures should ensure that code inputs use the correct data. ND's management of safety and quality assurance assessor considered

EDF and AREVA's approach to quality assurance, although not to quality assurance of shielding computational codes in particular (Ref. 127). Observations made by the TSC regarding key input data and assumptions used are summarised below.

168 Verification and validation of the computational codes used to perform the majority of shielding assessments were fit for purpose based on the documentation provided by EDF and AREVA discussed in the Section 4.3.1.4.1 above on Computational Codes (although supporting documentation for computational codes used to undertake activation calculations using TRIPOLI and DARWIN were not reviewed, as discussed in that Section). Advice on verification and validation is provided in paras 4.4.5 to 4.4.7 of the TAG on radiation shielding (Ref. 29).

169 For source data, the computational codes used in the UK EPR shield design were capable of adequately modelling the radiation source terms (discussed in Section 4.1 above) in sufficient detail for accurate dose rate prediction and were considered by the TSC to be acceptable.

170 Geometry modelling was used in calculations in a conservative manner so that resulting dose rates would also be conservative (Refs 75 and 82). Such modelling often omits plant items that could marginally reduce dose rates (e.g. stairways, structural supports). A number of shielding assessments used Computer Aided Design models (Ref. 82).

171 The responses to TQ-EPR-1227 (Ref. 8) and TQ-EPR-1228 (Ref. 8) provided information on the material cross-section libraries used. Although more recent updates are available for these cross-section libraries, the TSC considered that the libraries used were acceptable.

172 For dose points, room classifications were based on the calculated dose rate at 0.5m from the surface or a component or system from which the radiation was emitted (Ref. 67). Shielding assessment samples submitted by EDF and AREVA (the response to TQ-EPR-1231 (Ref. 8) and Ref. 68) provided evidence to show that dose rates were assessed at numerous locations taking account of all significant radiation sources within a room and / or radiation streaming from sources in adjacent rooms. The opinion of the TSC was that this should ensure a conservative calculation of the peak theoretical dose rates experienced by operators working within the room.

173 It is good practice in the UK to perform crosschecks of shielding calculations to provide confidence that the results are accurate, often by comparing the results to those calculated using an alternative analytical method (e.g. hand calculation, different computational code), or other comparable assessment and plant measurement data where appropriate. The statistical convergence for calculated dose rates quoted in documents provided by EDF and AREVA were all within acceptable limits, such as the evidence provided in the response to TQ-EPR-1353 (Ref. 8). EDF and AREVA indicated that results from MCNP calculations were crosschecked through comparisons with other codes, and validated by comparing measurements carried out inside Reactor Buildings of operating PWRs (response to TQ-EPR-1227 (Ref. 8). The TSC expected to see specific shielding examples where crosschecks had been performed, but although such crosschecks appeared not to have been undertaken, the TSC did not raise any concerns regarding the accuracy of computational shielding calculations.

4.3.1.5 Expectations – Calculation Methods

174 Advice on calculation methods is provided in Section 4.4, Calculation Methods, of the TAG on radiation shielding (Ref. 29).

4.3.1.6 Findings – Calculation Methods

175 In the opinion of the TSC, the computational codes used and their application in shielding assessments were appropriate. I concur with the opinion of the TSC from the evidence provided on calculation methods.

4.3.1.7 Assessment – Shielding Provisions

176 This part of my assessment reviewed the adequacy of shielding provisions to protect members of the public and workers.

4.3.1.7.1 Assessment – Protection of the Public from Direct Radiation

177 Protection of the public was achieved by ensuring that dose rates outside buildings were low, taking into account potential weak paths due to penetrations (e.g. doors, vents). The TSC undertook a high level review of the bulk shielding provisions for the UK EPR to verify that dose rates to members of the public were acceptable during all modes of normal operation (e.g. power operation, shutdown).

178 Chapter 11 of the Pre-construction Environment Report (PCER) (Ref. 83) stated that the annual dose uptake to members of the public living at 100m from a UK EPR unit was less than 6microSvy^{-1} based on dose rates at the exterior of the UK EPR being less than 1mSvy^{-1} (equivalent to 0.11microSvh^{-1}).

179 The response to TQ-EPR-1231 (Ref. 8) summarised results of shielding calculations which showed that dose rates at the exterior walls of the Reactor Building would be negligible. The TSC also reviewed and undertook basic crosscheck calculations of bulk shielding provisions for other buildings in the nuclear island based on plant layout drawings provided in the response to TQ-EPR-593 (Ref. 8). These crosscheck calculations showed that the bulk shielding provisions around all buildings in the nuclear island would reduce dose rates at the exterior surface of the facility to well below 1microSvh^{-1}. In addition, in the limited shielding assessment samples provided by EDF and AREVA in response to TQ-EPR-593 (Ref. 8) and TQ-EPR-1231 (Ref. 8), no obvious weak radiation paths from penetrations in the bulk shielding which could lead to much greater doses at the exterior of the facility were identified. The review undertaken by the TSC supported the statement in the PCER (Ref. 31) with regard to annual dose uptake to members of the public.

4.3.1.7.2 Expectations – Protection of the Public from Direct Radiation

180 Legislation on protection of members of the public from ionising radiation is IRR99 (Ref. 17), REPPIR (Ref. 18) and EPR10 (Ref. 19). Guidance on protection of members of the public from ionising radiation is in the ACOP and guidance to IRR99 (Ref. 21), in guidance to REPPIR (Ref. 22), in NT.1 Target 3 in the SAPs (Ref. 4), and in guidance on new NPPs from HPA RPD (Ref. 26). Guidance on radiation shielding is in T/AST/002 (Ref. 29). This legislation and guidance is discussed in Section [2.2], Standards and Criteria, of this assessment report.

4.3.1.7.3 Findings – Protection of the Public from Direct Radiation

181 In the opinion of the TSC, the bulk shielding provisions for the UK EPR to protect members of the public were adequate. I concur with the opinion of the TSC from the evidence provided on protection of the public from direct radiation.

182 As noted in Section 4.3.1.2.7 above, the response to the GDA issue, **GI-UKEPR-RP-01**, on radiological zoning and bulk shielding (see Section 4.2.4 above), will also provide further confirmatory information regarding the adequacy of materials used in bulk shielding.

4.3.1.7.4 Assessment – Protection of Workers from Direct Radiation

183 The approach taken by the TSC was to review the use of bulk shielding (e.g. shield walls, floors, doors) to verify that ambient dose rates were in line with the classification of areas criteria, and then to assess local shielding, penetrations and the use of temporary shielding on a sampling basis in cases where it could have a significant impact on dose accrual.

184 Local shielded items, such as glove boxes and sample transport trolleys, were not appropriate for consideration during GDA.

4.3.1.7.4.1 Bulk Shielding Provisions

185 As discussed in the section above on protection of the public from direct radiation, the bulk shielding provisions for the Reactor Building will ensure that dose rates at the surface of the Reactor Building will be negligible. However, no documents were provided to outline the radiological zoning classification scheme, or to show the calculated dose rates from shielding assessments for rooms in the UK EPR (see Section 4.2 above). In particular, the bulk shielding provisions could not be assessed to consider their contribution to the "two room concept" which allowed workers to enter the service space with the reactor at power (but not the equipment compartment). Therefore, the assessment of bulk shielding provisions were limited to samples of shielding assessments supplied by EDF and AREVA for the Reactor Building (Ref. 75, response to TQ-EPR-1353 (Ref. 8) and response to TQ-EPR-1231 (Ref. 8)), and for the -9.60m level of the Fuel Building (Ref. 82 and the response to TQ-EPR-1226 (Ref. 8)), supplemented with a general review of civil drawings for the UK EPR (response to TQ-EPR-593 (Ref. 8)).

186 For the assessment of bulk shielding provisions for the Reactor Building during Reactor Power, EDF and AREVA provided information outlining basic assumptions such as source terms, shielding geometry and calculation methods used to calculate dose rates within accessible areas (i.e. annular spaces and the service floor) (Refs 75 and 76), and details of neutron and gamma dose rates (responses to TQ-EPR-1353 (Ref. 8) and TQ-EPR-1231 (Ref. 8)). This evidence showed that calculated dose rates in annular spaces at the +5.15m level were less than the design criteria of 25microSvh^{-1} (Ref. 67). Peak dose rates on the service floor at the 19.50m level and within the fuel transport compartment were less than 10microSvh^{-1} and were considered as acceptable given the anticipated occupancy for these areas at full power operation (e.g. for access during 7 days before shutdown). The TSC also carried out a check on calculations using Atilla (a shielding code) which showed reasonable agreement with the neutron and gamma dose rates calculated at the +5.15m level annular spaces provided by EDF and AREVA.

187 For the assessment of priority rooms in the Fuel Building (-9.60m level), EDF and AREVA provided results of shielding calculations in a report that outlined modelling assumptions for features such as walls and shield doors segregating rooms containing sources of radiation (e.g. liquor tanks, vessels) from adjacent access corridors (Ref. 82). The report showed that the bulk thickness of the steel doors had been optimised to meet classification criteria in cold-side corridors. The report showed that dose rates in access corridors would be within the green zone criteria of less than 10microSvh^{-1}, and were often well below that criteria.

4.3.1.7.4.2 Local Shielding and Penetration Assessments

188 EDF and AREVA provided evidence of examples of optimisation of local shielding around penetrations in bulk shielding in the Reactor Building (Refs 75 and 82).

189 In the Reactor Building, neutron predictions around the primary coolant were very effective in limiting the spread of neutron radiation arising from the reactor vessel by using heat resistant neutron shielding to minimise the opening of the penetration through which the primary coolant pipe work passed, and the two significant gaps for radiation streaming were via the weak path through the insulation around the pipe work and via the path between the pressure vessel and the pool floor (Ref. 75 and the response to TQ-EPR-1353 (Ref. 8)).

190 In the priority rooms in the fuel building, local shielding was provided around wall penetrations to prevent radiation streaming into adjacent areas. The example showed optimisation of shielding to ensure that cold-side dose rates met radiological zoning criteria, and how potential weak paths were “designed out” by minimising gaps around doors and ensuring sufficient overlap of shielding (response to TQ-EPR-1226 (Ref. 8)). The TSC undertook a number of hand calculations and Atilla calculations which broadly confirmed the results of the shielding examples provided by EDF and AREVA.

4.3.1.7.4.3 Temporary Shielding Provisions

191 In general, temporary shielding is a matter for future operators, and the only areas relevant to GDA are assumptions regarding temporary shielding in dose estimations, and allocation of space for storage of temporary shielding.

192 The response to TQ-EPR-1225 (Ref. 8) showed that the collective dose for the UK EPR was based on the most recent / best operating French NPPs. This approach implicitly took account of temporary shielding used at these operational NPPs, but this approach was considered reasonable since UK operators were expected to implement similar good practices for dose reduction during operation and to use temporary shielding when necessary.

193 Even though it is not yet known how temporary shielding will be used and stored by future operators, it is appropriate for the UK EPR to provide adequate space for storage of temporary shielding, ideally in low dose rate areas near to where the activities will be undertaken. The information provided in the response to TQ-EPR-1225 (Ref. 8) showed that, where possible, temporary shielding had been replaced by permanent shielding (e.g. walls, labyrinths, local shielding), and space had been allocated for the storage of temporary shielding close to working areas within the UK EPR (except where areas for storage were strictly limited, e.g. Reactor Building annular spaces).

4.3.1.7.5 Expectations – Protection of Workers from Direct Radiation

194 Legislation on protection of workers from ionising radiation is IRR99 (Ref. 17) and REPPIR (Ref. 18). Guidance on protection of workers from ionising radiation is in the ACOP and guidance to IRR99 (Ref. 21), in guidance to REPPIR (Ref. 22) and in NT.1 Targets 1 and 2 in the SAPs (Ref. 4). Guidance on radiation shielding is in T/AST/002 (Ref. 29). This legislation and guidance is discussed in Section 2.2, Standards and Criteria, of this assessment report.

4.3.1.7.6 Findings – Protection of Workers from Direct Radiation

195 In the opinion of the TSC, the bulk shielding provisions and conservative dose rate predictions for the Reactor Building during power operation were acceptable given the anticipated occupancy requirements for these areas (e.g. for access during 7 days before shutdown), although evidence on shielding provisions and dose rates for other areas of the Reactor Building were not provided. EDF and AREVA provided a good example of how shielding provisions were assessed and optimised where necessary to ensure that bulk shielding and dose rates met radiological zoning criteria.

196 In the opinion of the TSC, the examples provided on local shielding and penetration assessments demonstrated that good design shielding practices and detailed calculation methods were used to ensure that potential weak paths in bulk shielding from penetrations and gaps would not breach radiological zoning criteria, and that the radiation protection guidelines (Ref. 67) were implemented in the design of the UK EPR.

197 In the opinion of the TSC, the final judgement on whether there is adequate space for the storage of temporary shielding can only be given once operators' requirements for temporary shielding are known. Therefore, the storage of temporary shielding should be reviewed by future operators once their requirements are known.

198 I concur with the opinion of the TSC from evidence provided on protection of workers from direct radiation with regard to bulk shielding provisions, local shielding and penetration assessments, and temporary shielding provision.

199 As noted in Sections 4.3.1.2.7 and 4.3.1.7.3 above, the response to the GDA issue, **GI-UKEPR-RP-01**, on radiological zoning and bulk shielding (see Section 4.2.4 above), will also provide further confirmatory information regarding the adequacy of materials used in bulk shielding.

200 I have identified a GDA assessment finding regarding temporary shielding as follows.

> ***AF-UKEPR-RP-03:*** *The licensee shall provide a report to identify areas where temporary shielding will be required for specific work activities and ensure there is adequate space for storage of such shielding when not in use, ideally in low dose rate areas near to the location where the work activities will be undertaken. This shall be complete before fuel on-site.*

4.3.1.8 Assessment - Dose Uptake

201 This part of my assessment considered how shielding assessments were used in the prediction and minimisation of dose uptake.

202 EDF and AREVA stated that the design target for collective exposure of any workers on the UK EPR was based on operational feedback from the French fleet to which the UK EPR dose optimisations were applied (response to TQ-EPR-1232, (Ref. 8)). EDF and

AREVA also supplied a number of references summarising dose optimisation studies for the most significant tasks undertaken during operation of the UK EPR (Refs 84 to 90). This information showed that the UK EPR design relied heavily on conservative shielding assessments to ensure that dose rates for rooms complied with radiological requirements. In contrast to this, dose predictions and optimisation were carried out using existing plant data that took account of design improvements to source term reduction and effective planning of work activities.

4.3.1.9 Expectations – Dose Uptake

203 It is best practice to use dose assessments to optimise shielding design with regard to radiation sources impacting on workstations to ensure that doses to workers for specific tasks are ALARP.

4.3.1.10 Findings – Dose Uptake

204 The UK EPR collective dose prediction and optimisation for workers was achieved without significant recourse to specific shielding assessment calculations. Although EDF and AREVA did not use dose assessments to optimise shielding design, in the opinion of the TSC, the approach taken by EDF and AREVA was acceptable. I concur with the opinion of the TSC from the evidence provided on dose uptake as it relates to shielding design.

4.3.2 Expectations – Summary for Shielding

205 The key pieces of legislation on the protection of workers and members of the public are IRR99 (Ref. 17), REPPIR (Ref. 18) and EPR10 (Ref. 19), and the key pieces of guidance are in the ACOP and guidance to IRR99 (Ref. 21) and in guidance to REPPIR (Ref. 22). In addition, guidance on BSLs and BSOs for workers on the site and people off the site during normal operation are in NT.1 Targets 1 and 2, and in NT.1 Target 3 in the SAPs (Ref. 4), respectively. Guidance on radiation shielding is available in RP.6 and paras 493 to 495 of the SAPs (Ref. 4), and in TAG T/AST/002 (Ref. 29).

4.3.3 Findings – Summary for Shielding

206 In the opinion of the TSC, the radiation protection guidelines (Ref. 67) and radiological zoning (Ref. 69) outlined good shielding practices and an appropriate scheme for designating areas with regard to radiological hazard from external radiation. The design basis data, calculation methods and computational codes used were adequate and were applied conservatively, which gave confidence that shielding provisions and predicted dose rates would also be conservative with regard to dose uptake from external radiation. The samples of EDF and AREVA's shielding assessments demonstrated how it had consistently used good shielding practices and appropriate calculation methods in the design and optimisation of shielding provisions.

207 The evidence provided confidence that shielding assessments for the UK EPR have been conducted using recognised shielding practices in a conservative manner comparable to methods typically used in the UK.

208 However, the radiological zoning and shielding summary data were not available for assessment during GDA. Since radiological zoning for restriction of exposure to ionising radiation of workers is fundamental to the design of the nuclear island, and bulk shielding

is inextricably linked with civil engineering aspects of that design, then radiological zoning and bulk shielding is a GDA Issue (**GI-UKEPR-RP-01**, see Annex 2). This is discussed in detail in Section 4.2 above on normal operations - designated areas (radiological classification of areas / radiological zoning).

209 There was one GDA assessment finding on temporary shielding, **AF-UKEPR-RP-03** (see Section 4.3.1.7.6 above).

210 I concur with the opinion of the TSC from the evidence provided on shielding.

4.4 Normal Operation – Contaminated Areas

4.4.1 Assessment – Contaminated Areas

211 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.

212 Contamination is a topic that spans both GDA and Phase 2 nuclear site licensing 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.

213 I assessed the approach of EDF and AREVA to contamination zoning (Ref. 71). I also sampled the locations of two work activities and examined the evidence provided by EDF and AREVA as examples of management of contamination control. These locations were the hot maintenance workshop (TQ-EPR-1166, Ref. 8) and air transfer system for movement of samples (TQ-EPR-1165, Ref. 8). I was supported in my assessment on contaminated areas by ND's TSC, TUV SUD (Ref. 51).

214 This topic is linked to Section 4.2 on designated areas and Section 4.7 on decontamination in this assessment report.

4.4.1.1 Assessment – Approach to Contamination Zoning

215 EDF and AREVA provided a document which was the cleanliness / waste design guide for the buildings FA3 EPR (Ref. 71). This document described the approach to be taken for contamination zoning for the rooms of the NPP which was linked with neutron fluxes in rooms, the presence or likelihood of presence of contaminating equipment or fluids in rooms, accessibility to access rooms during operation at power, and whether rooms had a contaminated fluid retention function (e.g. pool, sump). The need for air flow from non-contaminated areas through to the most heavily contaminated areas in order to minimise the spread of contamination is also described. The document outlined the need for changerooms, sub-changerooms, barriers, personal protective equipment and its storage at barriers, and how these requirements would change between operation at power and shutdowns. The types of equipment that would be required to measure contamination on workers, equipment and tools were also described.

216 As discussed in Section 4.2.2.2 above, I also raised TQ-EPR-1453 (Ref. 8) on access to controlled areas to request information / documentation on the layout of areas set aside

for accessing controlled areas, plus information / documentation showing possible locations for relevant health physics facilities for entering and exiting controlled areas. Such facilities would include changerooms. As already discussed, from the information provided in response to TQ-EPR-1453, it was not clear how the various rooms and pieces of equipment were linked together to form a logical path for entering and exiting the controlled area in order to minimise the spread of contamination. However, the level of detail provided was as expected at this stage in the design since such matters are the responsibility of future licensees. The key feature for the design is to provide sufficient space at suitable locations for the necessary rooms and pieces of equipment. I therefore raised **AF-UKEPR-RP-02** on access to controlled areas (see Section 4.2.4 above).

4.4.1.2 Expectations – Approach to Contamination Zoning

217 Guidance relevant to contamination zoning is included in RP.3, paras 485 and 486, and RP.4, paras 488-490, of the SAPs (Ref. 4), and in paras 4.6, 4.7, 4.9 and 4.10 of the TAG on Radiological Protection (Ref. 33). Detailed guidance on changerooms is provided in a nuclear industry code of practice on the design of changerooms (Ref. 99).

4.4.1.3 Findings – Approach to Contamination Zoning

218 In my opinion, from the evidence provided, the approach to contamination zoning was appropriate. The selection of equipment that would be required to measure contamination on workers, equipment and tools is a future operator's responsibility, and is therefore outside the scope of GDA and not part of my assessment.

4.4.1.4 Assessment – Hot Maintenance Workshop

219 The location identified for the hot maintenance workshop is on the +19.50m level in the Nuclear Auxiliary Building (response to TQ-EPR-1166, Ref. 8). This location is suitable since it would enable the easy transfer of tools, components, etc., from the Reactor Building and the Fuel Building to the hot maintenance workshop. The workshop also has sufficient space to facilitate the flexible use of machinery and tools for carrying out a range of tasks as required, and could accommodate the temporary erection of smaller workshops containing shielding (to reduce dose rates), contamination containment provisions (to minimise the spread of airborne contamination), ventilation (to provide an air flow from contamination free areas to contaminated areas, thereby minimising the risk of spread of airborne contamination) and mobile radiation monitoring equipment, as necessary.

220 The space allocated for the hot maintenance workshop and the areas immediately around it were sufficient to allow the establishment of separate contamination zones for tools, components and radioactive wastes. In addition, the tool storage facility is located adjacent to the hot workshop which could easily be accessed through two doors, and there is sufficient storage capacity for the storage of contamination-free equipment in close proximity to the hot maintenance workshop.

221 In addition to the hot maintenance workshop, the evidence showed that shielded glove boxes would be installed in specific rooms within the Nuclear Auxiliary Building for handling parts and components with high contamination levels or which emitted high dose rates. There are also rooms / cells in the Effluent Treatment Building fitted with shield / lead windows where concrete containers with high dose rates could be handled.

These shield / lead windows would enable crane operators to see handling operations during the work activities taking place in those rooms / cells.

4.4.1.5 Expectations – Hot Maintenance Workshop

222 Guidance relevant to the types of work activities that would take place in temporary work areas located in the hot maintenance workshop is included in RP.4, paras 488 to 490 of the SAPs (Ref. 4), and in paras 4.9 and 4.10 of the TAG on Radiological Protection (Ref. 33).

4.4.1.6 Findings – Hot Maintenance Workshop

223 In my opinion and in the opinion of the TSC, from the evidence provided, the hot maintenance workshop was suitably located and had sufficient space to allow the erection of temporary workshops etc. for work activities which could incorporate adequate safety features to minimise the risk of contamination spread and of exposure to direct radiation. The type of equipment that could be installed in the hot maintenance workshop is a future operator's responsibility, and therefore not part of my assessment.

4.4.1.7 Assessment – Sample Air Transfer System

224 The laboratories for handling radioactive samples were at level -9.60m in the Nuclear Auxiliary Building, level 0.00m in the Effluent Treatment Building and the first floor of the Operational Service Centre Building (which also had laboratories on the first floor for handling non-radioactive samples) (response to TQ-EPR-1165, Ref. 8). The response to this TQ also identified the route of the sample air transfer system, length of the piping involved, and speed of travel of samples.

225 To minimise the risk of leakage from sample containers, and thereby minimise the risk of spread of contamination within the piping of the air transfer system, liquid samples would be hermetically packaged before they were inserted into the sample transport containers. The sample containers were robust based on industrial experience, and in addition, contamination control measures would be put in place to ensure that the sample transport containers were contamination free (<4Bqcm^{-2} for beta / gamma emitters and <0.4Bqcm^{-2} for alpha emitters), both internally and externally, before being used. Therefore the risk of contaminating the transfer system's piping was minimised. Furthermore, all parts of the transfer system could be accessed manually, so that in case of contamination occurring, decontamination measures could be initiated and contaminated piping could either be decontaminated or replaced. The choice of containers and packaging of samples is the future operator's responsibility, and therefore not part of my assessment.

226 In the case of system failures, such as the loss of electrical power, piping damage or sample containers getting stuck in the piping, EDF and AREVA had provided adequate provisions to enable identification of the problem and reclamation of the sample container. Potential dose uptake for workers involved in recovery work was analysed by EDF and AREVA. The ambient dose rate for sample container recovery would be less than 20microSvh^{-1} at a distance of 0.5m from the piping. The contact dose rate at the surface of the sample container would be approx. 1mSvh^{-1} during sample recovery and transfer to an adapted transport container. EDF and AREVA judged this to be acceptable due to the short period of time a worker would be in close contact with the sample and the value of the dose limit for the hand being 500mSvy^{-1} (Ref. 17). These judgements appeared appropriate.

227 Similar systems for the transfer of samples around NPPs have been used reliably in German PWRs over several decades without encountering difficulties (Ref. 51), and similar systems have been used in the French fleet of NPPs without concerns being raised (response to TQ-EPR-1165, Ref. 8).

4.4.1.8 Expectations - Sample Air Transfer System

228 Guidance relevant to work activities involving the transfer of samples via an air transport system around the plant is included in RP.4, paras 488 to 490 of the SAPs (Ref. 4), and in paras 4.9 and 4.10 of the TAG on Radiological Protection (Ref. 33).

4.4.1.9 Findings - Sample Air Transfer System

229 In my opinion and in the opinion of the TSC, from the evidence provided, the sample air transfer system was an effective and dose saving way of transporting samples around the NPP.

4.4.2 Expectations – Summary for Contaminated Areas

230 Guidance relevant to contaminated areas is included in RP.3, paras 485 and 486, and RP.4, paras 488 to 490, of the SAPs (Ref. 4), and paras 4.6, 4.7, 4.9 and 4.10 of the TAG on Radiological Protection (Ref. 33). Detailed guidance on changerooms is provided in a nuclear industry code of practice on the design of changerooms (Ref. 99).

4.4.3 Findings – Summary for Contaminated Areas

231 In my opinion and in the opinion of the TSC, from the information provided, arrangements for contaminated areas were appropriate. In particular, the approach to contamination zoning is appropriate, the hot maintenance workshop was suitably located and had sufficient space for temporary workshops, the air transfer system is an effective and dose saving way of transporting samples around the NPP.

232 There were no GDA assessment findings associated with contaminated areas.

4.5 Normal Operation – Ventilation

4.5.1 Assessment – Ventilation

233 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 Reactor Building at power.
- Radiation exposures incurred during maintenance and testing.
- Control of naturally-occurring radon.

234 The assessment of ventilation systems within the UK EPR was undertaken by ND's mechanical engineering assessors, and the scope and findings of that assessment are described in the Step 4 Mechanical Engineering Assessment of the EDF and AREVA UK EPR Division 6 Assessment Report (Ref. 100). My assessment, therefore, sampled the radiological consequences arising from the ventilation system that allowed access into

the Reactor Building at power. I was supported in my assessment on ventilation systems by ND's TSC, TUV SUD (Ref. 51).

235 The key design feature of the Reactor Building was that it was built in two compartments: the equipment compartment (which enclosed the primary system elements); and the service space (where access was possible with the reactor at power if wearing basic protective clothing). The ventilation served to prevent radioactive airborne contamination from moving from the equipment compartment into the service space. Section 3 of Sub-Chapter 12.3 of the PCSR (Ref. 1) also outlined the purging process to allow entry into the equipment compartment for short periods of time. The purging process also allowed entry into the Reactor Building at all times during shutdown.

236 EDF and AREVA has undertaken a study to assess atmospheric contamination from gas compounds in their normal state (xenon), volatile liquid compounds able to evaporate and contaminate the airborne atmosphere (tritium), and compounds able to be volatilised as aerosols (iodine) (Ref. 101). During power generation, the two compartments are ventilated separately. The purpose of the study was to consider the impact of leakage from the equipment compartment into the service compartment. This was achieved by assessing the atmospheric contamination from radionuclides in both compartments during normal operation at full power in two cases: first, the case where there was no leakage between the two compartments; and secondly, in the case where there was some leakage between the two compartments. The results were expressed in terms of radionuclide concentrations at equilibrium, and the time taken to reach equilibrium. However, this study (Ref. 101) generated very conservative activity concentrations since the ventilation systems were designed to minimise the leakage between the two compartments by a difference in pressure (the ventilation maintained more pressure in the service compartment than the equipment compartment). In addition, the atmosphere in the equipment compartment was purified by the Reactor Building Internal Filtration System that re-circulated purified air. Calculated activity concentrations can be used to estimate their impact on workers carrying out work activities, but since the study (Ref. 101) generated very conservative activity concentrations, I did not assess the impact of these activity concentrations on workers.

237 Practical measures to control naturally-occurring radon gas are dependent on the geological characteristics of the site, and so were not considered in my assessment report.

4.5.2 Expectations – Ventilation

238 Guidance relevant to radiological protection aspects of ventilation is provided in RP.3, paras 485 to 486, RP.4, paras 488 to 490, and in paras 4.9 and 4.10 of the TAG on Radiological Protection (Ref. 33).

4.5.3 Findings - Ventilation

239 The assessment report on mechanical engineering (Ref. 100) summarises the assessment conclusions on ventilation with regard to mechanical engineering. In my opinion and in the opinion of the TSC, from the evidence provided, there was no reason to suppose that the ventilation was not suitable with regard to radiological protection.

240 Measuring the airborne concentrations of radionuclides in the Reactor Building before workers enter the service space with the reactor at power should be part of normal operational procedures. However, to follow on from the study on airborne concentrations

of radionuclides in the Reactor Building (Ref. 101), I have identified a GDA assessment finding regarding ventilation as follows.

> ***AF-UKEPR-RP-04:*** *The licensee shall provide an ALARP justification to establish the ventilation conditions in the equipment compartment and service space of the Reactor Building, including the airborne concentrations of radionuclides, under which workers may enter the service space with the reactor at power, in order to ensure that internal doses to workers are ALARP. This shall be complete before fuel on-site.*

4.6 Normal Operation – PRMS

4.6.1 Assessment - PRMS

241 My Step 4 Plan (Ref. 1) explained that my assessment of the PRMS 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.

242 My assessment was originally going to be carried out in liaison with ND's control and instrumentation assessors in order to assess control and instrumentation for monitoring direct radiation and contamination throughout the plant, including the ponds. However, these aspects of the PRMS, along with details of the monitoring equipment itself, were agreed as being out of scope of GDA. My assessment, therefore, sampled radiation exposures predicted to be incurred during maintenance and testing of installed radiation monitoring equipment.

243 I raised TQ-EPR-1198 (Ref. 8) on the location of radiation monitoring equipment. The PRMS was divided into two systems, namely the KRT system (for plant and systems monitoring for operation) and KRC system (for equipment to monitor the radioactivity present in working places for radiological protection for workers). The KRC system included mobile and installed radiation monitoring equipment, but the response to TQ-EPR-1198 dealt with the location of installed radiation monitoring equipment only (since mobile radiation monitoring equipment would be the responsibility of future operators). The KRT system comprised only installed radiation monitoring equipment.

244 The response to TQ-EPR-1198 (Ref. 8) provided unique identifiers for installed radiation monitoring equipment of the KRT and KRC systems, and evidence on the types of radiation measured, the location of the sensors, the location of the monitors, the building within the nuclear island, and whether the measurements were for routine operation and / or accident conditions. The evidence also indicated the locations where radiological zoning would change with the operational state of the plant, e.g. when the reactor was at power, or during reactor shutdown. The alarms for all of the fixed radiation monitoring equipment were provided locally and in the main control room of the NPP.

245 The response to TQ-EPR-1198 (Ref. 8) showed that specific requirements were placed on the equipment in the KRC and KRT systems in terms of the mean time between failure of the equipment (which had to be less than 20,000 hours), and the mean time to repair (which had to be more than 30 minutes). The equipment in the KRC system was not safety classified, but I did not assess whether or not this was appropriate. The safety classification scheme for equipment in the KRT system was presented for the FA3 EPR,

but this was not assessed since the control and instrumentation aspects of the KRC and KRT systems may be assessed during Phase 2 nuclear site licensing, as necessary.

246 EDF and AREVA provided information on the expected dose rates at the installation points of the radiation monitoring equipment to measure radiation from gamma-rays, neutrons, aerosols, iodine and noble gases, and on the frequency with which workers had to replenish filters on that equipment (for measuring aerosols, iodine and noble gases). Equipment requiring frequent attention was located, SFAIRP, in the lower dose rate areas.

247 In the Reactor Building, the KRC system installed equipment was in rooms where workers had access, and the radiation sensors and the monitors and alarms were in the same room. In these cases, the ambient dose rates in the rooms would be less than 25microSvh^{-1} gamma radiation and less than 2.5microSvh^{-1} neutron radiation. Alarms were also installed in the main control room and at the main entrance to the Reactor Building.

248 In the Nuclear Auxiliary Building, and in some rooms in the Effluent Treatment Building, the KRC system installed equipment was set up to measure radioactivity on filters. With such equipment, it is important to minimise the distance between the sampling point and the filter so that samples are not degraded before they reach the filter. Therefore, the filters and monitors were located as near to the sensors as possible, which in these cases were in the rooms above those where the sensors were installed. From shielding provisions and source terms, the dose rates in the rooms above were not expected to exceed 26microSvh^{-1}. Consideration had been given to locating the filters and monitors in lower dose rate areas nearby (instead of in the rooms above), but this option had been discarded since it would have resulted in over-crowding in those lower dose rates areas, which in these cases were in corridors. Locating the filters and monitors in the corridors would also have resulted in additional penetrations for the cables between the sensors, filters and monitors, which may have created weak radiation paths into those corridors.

249 In other rooms of the Effluent Treatment Building, the KRC system installed equipment was present to measure increases in gamma-rays and release of aerosols from faults arising during waste conditioning and storage of concrete drums. Since the distance between the sensors and the filters and monitors needed to be minimised, they were all installed in the same room. Maintenance of this equipment would take place when work activities that may generate aerosols were not taking place, and additional temporary shielding could be used to reduce doses from external radiation.

250 No information was provided on the location of sensors, filters and monitors for the KRT system installed equipment with regard to minimising doses to workers taking measurements or maintaining such equipment, but there was no reason to suppose that the same approach for the reduction of doses to workers undertaking such work activities would not be taken.

251 The radiological zoning scheme for the UK EPR would not be finalised until Phase 2 nuclear site licensing since future operators may have different requirements than the FA3 EPR (see Section 4.2 of this assessment report). Therefore, the exact location of installed radiation monitors cannot be finalised until the radiological zoning classification scheme has also been finalised.

4.6.2 Expectations - PRMS

252 Guidance relevant to radiological protection aspects of plant radiation monitoring systems is included in RP.2, paras 482 to 484 of the SAPs (Ref. 4), and in para. 4.5 of the TAG on Radiological Protection (Ref. 33).

4.6.3 Findings - PRMS

253 In my opinion, from the evidence provided, the approach taken to identify the location of installed radiation monitoring equipment, and in particular, the location of sensors, filters and monitors, was adequate with regard to minimising doses to workers carrying out maintenance and testing of that equipment. The factors taken into account by EDF and AREVA should help to ensure that doses to those workers would be ALARP.

254 I have identified two GDA assessment findings on the PRMS as follows.

> ***AF-UKEPR-RP-05:*** *The future licensee shall provide a report to demonstrate that the control and instrumentation aspects of the installed radiation monitoring equipment of the KRC and KRT systems of the PRMS are adequate. This shall be complete before mechanical, electrical and control and instrumentation safety systems inactive commissioning.*
>
> ***AF-UKEPR-RP-06:*** *The future licensee shall provide a report to demonstrate that the planned location of the installed radiation monitoring equipment of the KRC and KRT systems of the PRMS are appropriate and take account of the final radiological zoning classification scheme with regard to ensuring that radiation exposures received by workers whilst taking measurements or maintaining or testing such equipment are ALARP. This shall be complete before mechanical, electrical and control and instrumentation safety systems inactive commissioning.*

4.7 Normal Operation – Decontamination

4.7.1 Assessment – Decontamination

255 My Step 3 Radiological Protection Assessment of the EDF and AREVA UK EPR Assessment Report (Ref. 6) concluded that claims for decontamination provision were not fully appropriate in that I expected more decontamination facilities to be located within the NPP, such as for bringing items out of controlled areas (although monitoring and decontamination of small items may be associated with changerooms). I therefore stated that the claims, arguments that underpin those claims, and evidence to substantiate those arguments, would be secured and assessed during Step 4.

256 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.

257 During Step 4, ND and the Environment Agency were considering EDF and AREVA's plans for decommissioning a UK EPR, 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, RO-UKEPR-77, that covered

both operations and decommissioning (Ref. 9). ND and the Environment Agency also raised a RO on decommissioning, RO-UKEPR-67 (Ref. 9).

258 In preparing RO-UKEPR-77 (Ref. 9), ND and the Environment Agency again reviewed the information on decontamination that was contained within the PCSR (Ref. 11) and PCER (Ref. 83) and identified a number of aspects of decontamination that the GDA submission needed to address. The PCSR (Ref. 1), PCER (Ref. 83) and supporting documents did contain some outline information on decontamination. However, it was not possible to identify a number of key features within the documents, and these are presented below.

259 First, we were not able to identify detail on the baseline decontamination strategy for the UK EPR during operations and maintenance, including the following.

- 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 were 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 were not already included within the GDA design.
- Clarity on what level of automation would be involved.

260 Secondly, we were not able to identify detail on the baseline decontamination strategy for a UK EPR during Post Operational Clean Out (POCO) and decommissioning, including the following.

- 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.

261 Thirdly, we were not able to verify that the wastes arising from decontamination operations had been considered, including the following.

- Detail on how decontamination waste arisings had been minimised through both the design and the operational, maintenance and decommissioning philosophies developed for a UK EPR.
- Consistency with the environmental submissions, e.g. Chapters 3, 6, 8 and 11 of the PCER (Ref. 83), and the Radioactive Waste Management Directorate disposability assessment (Ref. 102).
- How knowledge will be managed over the lifecycle, particularly knowledge pertinent to decontamination.

262 Therefore, we raised RO-UKEPR-77 to address the shortfalls described above, and we raised five ROAs associated with the RO (Ref. 9) on the topics listed below.

- ROA-UKEPR-77.A1: Decontamination during operations and maintenance.

- ROA-UKEPR-77.A2: Decontamination during POCO and decommissioning.
- ROA-UKEPR-77.A3: Laundry facilities.
- ROA-UKEPR-77.A4: Decontamination wastes.
- ROA-UKEPR-77.A5: Knowledge management.

263 EDF and AREVA's responses to RO-UKEPR-77 on decontamination (Ref. 9) and RO-UKEPR-67 on decommissioning (Ref. 9) are in Refs 103, 104 and 105, respectively. These responses were assessed by ND's TSC, REACT Engineering Limited, and the TSC's findings are reported in Ref. 54. EDF and AREVA subsequently provided a revised version of Ref. 105 on decommissioning that took account of comments from ND (Ref. 106). The assessment of EDF and AREVA's response to RO-UKEPR-67 on decommissioning is discussed in the Step 4 assessment report on radioactive waste and decommissioning (Ref. 56).

264 ROA-UKEPR-77.A1 (Ref. 9) requested EDF and AREVA to provide its baseline decontamination strategy for a UK EPR during operations and maintenance. The use of decontamination techniques may have a significant impact on other aspects of plant operation, such as the methodologies adopted (manual or remote), shielding and containment requirements, and eventual decommissioning techniques used. The decontamination techniques may also have a significant effect on the operational and decommissioning waste routes, including disposal of decontamination wastes (which were considered further under ROA A4). ND and the Environment Agency therefore expected to see details on the overall decontamination strategy and techniques during operations and maintenance, which should include techniques for both individual items and complete systems. ROA-UKEPR-77.A1 (Ref. 9) requested EDF and AREVA's response to take account of the following matters.

- Provide detail on the predicted decontamination requirements during operations and maintenance, including any key plant areas, systems, structures, components, etc., which may regularly require decontamination during operations and maintenance.
- Provide a baseline decontamination strategy / philosophy which could be adopted by future operators, including whether decontamination could be done in-situ or at a designated location(s).
- Provide detail on any design features to support decontamination.
- Provide detail on any decontamination systems which were included in the GDA design for deployment during operations and maintenance.
- Provide clarity on the decontamination systems and techniques which could be used by future operators in those areas where decontamination systems were not already included within the GDA design.
- Provide clarity on what level of automation would be involved.

265 ROA-UKEPR-77.A2 (Ref. 9) requested EDF and AREVA to provide its baseline decontamination strategy for a UK EPR 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-UKEPR-77.A2 (Ref. 9) requested EDF and AREVA's response to take account of the following matters.

- Provide detail on the predicted decontamination requirements during POCO and decommissioning, including any key plant areas, systems, structures, components, etc., which may require POCO or decontamination as part of decommissioning.
- Provide a baseline decontamination and POCO strategy / philosophy which could be adopted by the site operator, including whether decontamination would be done in-situ or at a designated location(s).
- Provide detail on the baseline decontamination systems and techniques assumed in the GDA design, including details on any enabling design features provided, e.g. material selection, space / layout, and connection points.

266 ROA-UKEPR-77.A3 (Ref. 9) requested EDF and AREVA 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. However, since laundering could be done in dedicated on-site laundry facilities or centralised off-site facilities, and since selection of the means of laundering would be the decision of future operators, towards the end of the assessment period it was agreed that laundry facilities should be outside the scope of GDA (this is covered in the Step 4 radioactive waste and decommissioning assessment report (Ref. 56)). Discharges from laundry facilities were within in the scope of GDA and were assessed by the Environment Agency (Ref. 117).

267 ROA-UKEPR-77.A4 (Ref. 9) requested EDF and AREVA to provide information regarding decontamination wastes. The assessment of EDF and AREVA's response was undertaken by the Environment Agency (Ref. 117).

268 ROA-UKEPR-77.A5 (Ref. 9) requested EDF and AREVA to provide information regarding knowledge management. EDF and AREVA's response (Refs 104 and 105) was covered by its response to ROA-UKEPR-67.A8 (Ref. 9). The assessment of EDF and AREVA's response was undertaken by ND's radioactive waste and decommissioning assessors (Ref. 56).

269 The TSC concluded that, in general, the response to RO-UKEPR-77 on decontamination addressed all aspects of the RO (Ref. 54). Some areas were left to future licensees, but this was appropriate. The information also made some comparisons with existing sites to help underpin the strategies / techniques which were proposed. The key factor was that the predicted decontamination requirements were identified and that a baseline approach was set out for each type of decontamination likely to be required. AMEC's literature review of radiological protection and radioactive waste and decommissioning practices during the last 10 years of normal operation of PWRs (Ref. 49) provided useful benchmark information.

270 A summary of the TSC's assessment of EDF and AREVA's response to the ROA on decontamination during operations and maintenance (ROA A1) is provided below.

- The baseline strategy highlighted in-situ decontamination as the preferred philosophy, although it recognised that many items would need to be transferred to dedicated locations for various logistical reasons. A large workshop area was outlined for this purpose, along with an overview of transfer routes from all main areas of the nuclear island, and this was one of the key areas that gave confidence in the ability to decontaminate a UK EPR.
- Decontamination techniques were set out for large items, small items, plant areas, ponds / pits, decontamination in the event of accidents, and decontamination of

workers. The information was sometimes limited, but was sufficient for GDA. Optional systems that future operators may wish to consider were also outlined, but their application to the UK EPR was not assessed since this was not necessary for GDA.

- No automated systems were included in the design (e.g. for pool cleaning) and this was appropriate for GDA, although future operators may wish to use automated systems and remote handling devices to reduce radiation doses to workers.
- The information was not clear on how in-situ decontamination of some items could be carried out (in-situ was identified by EDF and AREVA as the preferred approach where possible). The in-situ decontamination techniques were set out for the primary circuit but less information was provided for other systems. It was also unclear how the CORD UV system would actually be deployed for the primary circuit. Such information need not necessarily be provided as part of GDA because the selection of decontamination systems, strategies and techniques will be the choice of future operators. Therefore, future operators would need to develop their own site-specific strategies, systems and techniques for decontamination during operations and maintenance, whilst taking account of the contamination zoning of the NPP. I have captured this requirement in **AF-UKEPR-RP-07** below (see Section 4.7.3).

271 A summary of the TSC's assessment of EDF and AREVA's response to the ROA on decontamination during POCO and decommissioning (ROA A2) is provided below.

- The baseline decontamination strategy for POCO and decommissioning was not specified as such, but it could be inferred. The strategy included a full system decontamination approach for the primary circuit using the CORD UV process, and provided examples of where localised decontamination would be required using other techniques for auxiliary systems, structures, components, and cabling.
- The strategy included re-use of existing facilities where practicable, but in some cases it was not completely clear which decontamination systems were included in the baseline strategy. As discussed above with regard to decontamination during operations and maintenance, it would have been helpful to gain an understanding of the logistical requirements to deploy the CORD UV system on the primary circuit, but again, this would need to be resolved by future operators rather than within GDA.

272 With regard to laundry facilities (ROA A3), the baseline GDA submission assumed the use of off-site facilities and so no design for an on-site laundry was included, although the response did include the discharges for a laundry and outlined the key considerations and space required should an on-site laundry be provided. Future licensees will need to identify their strategy for laundry provision which may include the design of an on-site laundry.

273 The response also considered alignment with the disposability assessments and environmental submissions (ROA A4). I did not review these topic areas since they are covered in the Environment Agency's assessment report on disposability (Ref. 128) and in ND's Step 4 radioactive waste and decommissioning assessment report (Ref. 56).

274 As already explained, the response on knowledge management was incorporated into the response to ROA A8 of RO-UKEPR-67 (Ref. 105) and was reviewed by the Environment Agency's assessors (Ref. 117) and ND's radioactive waste and decommissioning assessors (Ref. 56).

4.7.2 Expectations - Decontamination

275 Guidance relevant to decontamination is included in RP.5, paras 491 to 492 of the SAPs (Ref. 4), and in paras 4.11 and 4.12 of the TAG on Radiological Protection (Ref. 33).

4.7.3 Findings - Decontamination

276 In the opinion of the TSC, from the evidence provided, there was sufficient confidence in the ability to decontaminate a UK EPR during POCO and decommissioning. There was also confidence in the ability to decontaminate a UK EPR during operations and maintenance, although additional evidence could have been provided to increase this confidence level. Nevertheless, no showstoppers or significant outstanding concerns were identified.

277 I concur with the opinion of the TSC from the evidence provided on decontamination during operations and maintenance, and during POCO and decommissioning.

278 I have identified GDA assessment findings regarding decontamination as follows.

> ***AF-UKEPR-RP-07:*** *The licensee shall provide a report to demonstrate its site-specific strategies, systems and techniques for decontamination during operations and maintenance, and during POCO and decommissioning, whilst taking account of the contamination zoning of the NPP. This shall be complete before fuel on-site.*
>
> ***AF-UKEPR-RP-08:*** *The licensee shall provide a report to demonstrate its strategy for laundry provision, and if that strategy includes an on-site laundry, then that laundry shall be designed such that its location and containment integrity will ensure that doses to workers are ALARP. This shall be complete before first structural concrete.*

4.8 Normal Operation – Optimisation for Work Activities (Including Fuel Route)

4.8.1 Assessment – Optimisation for Work Activities (Including Fuel Route)

279 To begin my assessment, I raised TQ-EPR-596 (Ref. 8) that requested information / documentation on high dose work activities (including the fuel route, maintenance activities and jumpers) 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 matters as indicated below which were taken from my Step 4 Plan (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 dose for work activities.
- Substantiation of improvements leading to reductions in estimated doses.
- Doses to employees not working with ionising radiations.
- Alternative means of carrying out high dose work activities.
- Management of exposures in timeframes measured in seconds.
- Demonstration of ALARP.

280 I was supported in my assessment on optimisation for work activities by ND's TSC, TUV SUD (Ref. 51). The information and references provided in response to this TQ-EPR-596 (Ref. 8) were used as the groundwork for the assessment. The general areas considered within this part of my assessment were as follows.

- Collective Doses (Basis of Collective Dose Estimate, Calculation of Collective Dose Estimate, Annual Collective Dose).
- Highest Dose Work Activities (Steam Generator Inspections and Maintenance, Fitting and Removing Insulation).
- Work Activities Associated with Fuel Transfer.

4.8.1.1 Assessment - Collective Doses

4.8.1.1.1 Assessment - Basis of Collective Dose Estimate

281 The collective dose estimate for the UK EPR was discussed in Sub-Chapter 12.4 of the PCSR (Ref. 11). The collective dose estimate was developed from collecting dose uptake statistics from 1300 MWe NPPs (Ref. 91), starting with information from the NCAD codes from the best operating NPPs from the French fleet (NCAD codes - a New Framework for Dosimetry Analysis – NDAC codes were unique and corresponded both to an elementary activity performed by a worker in a controlled area and to the access code for this worker in the controlled area). Dose uptake data associated with design features and / or tasks which had the greatest similarity to the UK EPR design would be of greatest value. The information from the codes was supplemented with dose data for maintenance, and supplemented further with data from German nuclear facilities where work activities associated with the UK EPR were similar to those of German Konvoi reactors (e.g. work activities involving aeroballs).

4.8.1.1.2 Expectations - Basis of Collective Dose Estimate

282 International guidance on radiological protection for NPPs from the IAEA (Ref. 42) advises that, in setting design targets for occupational radiation doses, account should be taken of experience at relevant plants that have a good operating record in terms of radiological protection.

4.8.1.1.3 Findings - Basis of Collective Dose Estimate

283 In my opinion and in the opinion of the TSC, from the evidence provided, EDF and AREVA provided sufficient information to show that the calculation of the estimated annual collective dose was based on a number of French and German power plants that had the best operating records in terms of radiological protection (Ref. 11), therefore the basis for collective dose estimation was adequate in that it was undertaken in accordance with IAEA international guidance (Ref. 42).

4.8.1.1.4 Assessment – Calculation of Collective Dose Estimate

284 EDF and AREVA established a reference annual collective dose for the UK EPR based on operational feedback data from PWRs. EDF and AREVA then identified a number of design improvements for the UK EPR that would result in an estimated reduction in the annual collective dose for the UK EPR (Ref. 11).

285 Sub-Chapter 12.4 of the PCSR (Ref. 11) reported that the calculated reference dose was 0.448personSvy^{-1} per unit (per reactor) which was determined from recent statistical values for the best performing French units.

286 This calculated reference dose was established using a number of factors / assumptions as follows.

- Use of the best up-to-date dose statistics from the recent French units (2001 to 2003) of the P4 and N4 series (Ref. 91).
- UK EPR fuel cycle of 18 months.
- Doses were averaged over ten years, where the ten-year outage cycle consisted of NRO-ROO-NRO-ROO-NRO-ISIO, where NRO was a normal refuelling outage, ROO was a refuelling only outage, and ISIO was an in-service inspection outage (which was carried out every ten years). Each type of outage was associated with different dose values depending on the work activities carried out during those outages.

287 The UK EPR collective dose estimate was calculated by taking into account factors that would change the dose compared to the calculated reference dose (Refs 11 and 91). There were four main factors that influenced the calculation of the collective dose estimate as follows.

- First, source term and dose rate optimisation. This was achieved by a number of means, which were optimising / reducing the use of Stellites^{TM} (to reduce dose rates from activated crud in the primary circuit, see Section 4.1 of this report), pressuriser developments (e.g. installation of a floor separating the spray and discharge systems at the pressuriser dome level to reduce the average dose rate in the safety valves area), installation optimisation (e.g. inclusion of a shielded area dedicated to storing the pressure vessel head in the Reactor Building), and measures to remove hot spots in the design (e.g. elimination of pipe connections using socket welds on pipework carrying radioactive fluids).
- Secondly, optimising the amount of exposed work. This was achieved by a number of means, including increasing primary and secondary personway diameters (to improve access thereby reducing exposures), inclusion of a cylindrical section on steam generator channel heads (to provide better access to peripheral tubes thereby reducing exposures), installing the reactor vessel head heat installation as a removable single unit (to reduce handling thereby reducing exposures), and installing modular maintenance valves (to avoid some maintenance tasks thereby reducing the need for exposures to maintenance workers).
- Thirdly, a number of design options may impact on dose assessments, such as electro-polishing the steam generator channel heads, but since such work had not been finalised, any impacts from such design options on dose uptake had not been assessed.
- Fourthly, a specific feature of in-house intervention. This was to improve the accessibility of the UK EPR Reactor Building while the reactor was in operation, specifically for seven days before shutdown (to prepare for an outage) and three days after start-up (to complete the work following an outage). This was achieved by developing the two-room concept where there were two compartments within the Reactor Building (Ref. 107). These compartments were the equipment compartment (where dose rates and levels of airborne radioactivity were too high for worker entry) and the service space (where doses rates were below 25microSvh^{-1} gamma and 2.5microSvh^{-1} neutron radiation, and the levels of airborne radioactivity were

sufficiently low to allow workers to enter without the need for wearing respiratory protective equipment). I have identified a related assessment finding on ventilation (**AF-UKEPR-RP-04**, see Section 4.5 above) to establish the conditions under which internal doses to workers entering the service space during operation at power would be ALARP, and for those conditions to be established before workers entered the service space with the reactor at power for the first time.

288 The UK EPR dose estimate was further refined and improved by taking into account improvements in dose optimisation of seven work activities that resulted in fifty per cent of the collective dose. These work activities were thermal insulation operations, worksite logistics, valve activities, steam generator (SG) worksite, worksite for opening and closing the reactor vessel, fuel posting out worksite, and waste treatment operations.

289 In my Step 3 assessment report (Ref. 6), I reported that the estimated doses for these work activities in Sub-Chapter 12.4 of the PCSR (Ref. 11) were different from those in the supporting documentation. In that report I explained that the first optimisation step had assumed a 10% reduction in source term, the second step had assumed a further 15% due to the removal of Stellite™ from materials in contact with the primary circuit, and the third and final optimisation step for GDA was ongoing and would take account of any further planned improvements.

290 TQ-EPR-596 (Ref. 8) requested evidence of substantiation of improvements, including the 10% and 15% reductions in estimated doses during successive steps of the optimisation process, and this was provided in the references listed below. I did not review all of these documents in detail during my Step 4 assessment, but instead reviewed certain ones in detail during sampling and in-depth technical assessment of some of these work activities.

- Thermal insulation operations (Ref. 89).
- Worksite logistics (Ref. 90).
- Valve activities (Ref. 84).
- SG worksite (Ref. 85).
- Worksite for opening and closing the reactor vessel (Ref. 86).
- Fuel posting out worksite (Ref. 87).
- Waste treatment operations (the original reference submitted as supporting documentation during the Step 3 assessment covered steps 1 and 2 of the optimisation process, Ref. 88).

4.8.1.1.5 Expectations – Calculation of Collective Dose Estimate

291 Guidance on strategies to restrict radiation exposure is in para. 479 of the SAPs (Ref. 4). This includes establishing a hierarchy of control measures, as discussed in paras 478 and 480 of the SAPs (Ref. 4), which in turn refer to guidance in L121 (Ref. 21) regarding IRR99 (Ref. 17). Guidance on strategies to restrict exposure is also in paras 4.1 and 4.2 of the TAG on Radiological Protection (Ref. 33), and guidance on the hierarchy of control measures is also in para. 4.3 of that TAG.

4.8.1.1.6 Findings – Calculation of Collective Dose Estimate

292 In my opinion and in the opinion of the TSC, from the evidence provided, the strategy used to calculate the collective dose estimate, which involved successive stages of optimisation, was suitable and sufficient.

4.8.1.1.7 Assessment – Annual Collective Dose

293 The calculated annual collective dose (EDPO) for workers in the PCSR was 345person-mSvy^{-1} per unit (Ref. 11). This value was derived from the total collective dose over a period of ten years (which included NRO, ROO and ISIO outages with estimated outage doses of 390person-mSvy^{-1}, 197person-mSvy^{-1} and 996person-mSvy^{-1}, respectively, plus an estimated operational dose per year of 89person-mSvy^{-1}, assuming an 18-month campaign cycle, averaged over ten years). Following completion of the dose optimisation studies, the calculated annual collective dose (EDPO) for workers in the cPCSR was 340person-mSvy^{-1} per unit (Ref. 13). This value was also derived from the total collective dose over a period of ten years (which included NRO, ROO and ISIO outages with estimated outage doses of 372person-mSvy^{-1}, 200person-mSvy^{-1} and 972person-mSvy^{-1}, respectively, plus an estimated operational dose per year of 91person-mSvy^{-1}, assuming an 18-month campaign cycle, averaged over ten years).

4.8.1.1.8 Expectations – Annual Collective Dose

294 Guidance in para. 479 of the SAPs (Ref. 4) advises that an important element of optimisation of protection is that the collective effective dose to people on-site, as a result of the operation of the nuclear facility, should be kept ALARP. To fulfil this expectation and the requirements of restriction of exposure in regulation 8 of IRR99 (Ref. 17), I would expect that the annual collective dose should be comparable with the annual collective dose of the best operational existing plants.

295 The NEA report on the design of NPPs published in 2010 (Ref. 43) indicated that the operational average annual collective dose for all PWRs between 1997 and 2007 fell from approximately 1225person-mSv to approximately 750person-mSv (Figure 2 of Ref. 43). According to ISOE, the worldwide operational average annual collective dose for all PWRs in 2008 was 690person-mSv (Ref. 95). In contrast to this, the NEA report (Ref. 43) indicated that the operational average annual collective dose for advanced PWRs between 1997 and 2007 was between 200person-mSv and 350person-mSv with no obvious trend in doses over successive years (Figure 2 of Ref. 43). In addition, the NEA report (Ref. 43) suggested that, based on ISOE data, the operational annual collective dose benchmarks for new advanced PWRs could be in the order of 250person-mSv.

4.8.1.1.9 Findings – Annual Collective Dose

296 In my opinion and in the opinion of the TSC, from the evidence provided, the calculated average annual collective dose (EDPO) of 345person-mSv for the UK EPR (Ref. 11), which was superseded (following completion of the dose optimisation studies) by an EDPO of 0.34person-mSv (Ref.13), is consistent with the operational average annual collective dose of advanced PWRs. Experience shows that calculated doses are almost always conservative when compared with operational doses. In this case, the calculated average annual collective dose for the UK EPR was within the range of the operational average dose for operational advanced PWRs between 1997 and 2007. Although this was at the upper end of the range, this was not entirely unexpected since the collective

dose for the UK EPR was a planning estimate, whereas the collective doses for the operational advanced PWRs were from operational data which are almost always lower than the preceding calculated doses. Therefore, in my opinion and in the opinion of the TSC, the calculated average annual collective dose of 345person-mSv is acceptable, and the future operational average annual collective dose of the UK EPR is expected to be lower than this value.

4.8.1.2 Expectations – Summary for Collective Doses

297 Guidance relevant to collective doses is available in international guidance on the design of new NPPs published by the IAEA (Ref. 42) and NEA (Ref. 43), in the ISOE database of 2008 (Ref. 95), in IRR99 (Ref. 17) and its ACOP and guidance (Ref. 21), in ND's SAPs (Ref. 4), and in ND's TAG on Radiological Protection (Ref. 33).

4.8.1.3 Assessment – Summary for Collective Doses

298 In my opinion and in the opinion of the TSC, from the evidence provided, EDF and AREVA had provided sufficient information to show that the basis for the collective dose estimate was adequate, the strategy used to calculate the collective dose estimate was suitable and sufficient, the calculated average annual collective dose was acceptable and the future operational annual collective dose was expected to be lower than this value.

4.8.1.4 Assessment – Samples of High Dose Work Activities

299 The purpose of my assessment was to consider two examples from the list of the high dose work activities that represented 50% of the total collective dose (see Section 4.8.1.1.4 above). The work activities that I selected were SG inspections and maintenance, and fitting and removing insulation. I also considered radiological protection aspects of waste treatment operations in Section 4.9 below.

4.8.1.4.1 Assessment – SG Inspections and Maintenance

300 EDF and AREVA had made a number of improvements that would reduce doses to workers carrying out high dose work activities associated with SGs (Ref. 85), and some of these involved design changes. In addition, EDF and AREVA were carrying out a number of feasibility studies designed to reduce exposures of these workers even further, and these studies were ongoing (Ref. 85).

301 My assessment considered these high level improvements and studies, then sampled some high dose work activities and did an in-depth technical assessment of some specific SG inspection and maintenance work activities, namely tube plugging and testing, non-destructive testing (NDT) and visual inspection.

302 The annual collective reference dose for SG preparation and tests in the UK EPR during the first phase of the dose optimisation studies was 40.1person-mSvy^{-1}, and following completion of the dose optimisation studies, the annual collective reference dose for SG preparation and tests was 44.8person-mSvy^{-1} (Ref. 85). The annual collective dose for one year was the average over a period of ten years, during which there would be a pattern of outages that included three NROs, two ROOs and one ISIO, as described in Section 4.8.1.1.4 above. These work activities clearly had a large impact on collective and individual doses. The reference dose for SG preparation and tests (40.1person-mSvy^{-1}) was then reduced sequentially by taking credit for improvements. The annual

collective dose after taking account of improvements that had been implemented was 30.2person-mSvy^{-1}. For improvements that had been identified but not yet implemented, the annual collective dose fell to 24.7person-mSvy^{-1}, and for improvements that were still being studied the annual collective dose was predicted to fall even further to 22.6person-mSvy^{-1}, which if achieved would be a dose saving of approx. 44%.

303 The purpose of this part of my assessment was to consider the evidence to support these predictions. Ref. 85 outlined eleven substantial tasks associated with SG preparation and tests, and provided the percentage of the total dose that each of these tasks comprised for each of the three types of outage. The eleven tasks were as follows. (The first phase of the optimisation studies only estimated the first five of these tasks.)

- Preparation on reactor coolant side.
- Inspections on reactor coolant side.
- Preparation on secondary system side.
- Lancing.
- Inspections on secondary system side.
- Repairs on reactor coolant side.
- Repairs on secondary system side.
- Secondary system hydraulic test.
- SG welds liquid penetrant testing (concerns reactor coolant system).
- SG connection radiographic testing (concerns the reactor coolant).
- SG ultrasonic testing (concerns the secondary system).

4.8.1.4.1.1 UK EPR Improvements for SG Inspection and Maintenance

304 There were four UK EPR improvements that may impact of doses received by workers carrying out SG inspection and maintenance work activities (Ref. 85). The obvious outcome of some of these improvements was to reduce outage times, but there were also clear benefits in terms of dose optimisation.

305 First, there was the reduction in the source term. This has already been discussed in Section 4.1 above with regard to management of information on the source term (RO-UKEPR-73, Ref. 9) and reductions in the source term with regard to cobalt, silver and antimony reduction. In summary, the management of information on source terms and reductions in source terms were considered suitable and sufficient.

306 Secondly, there were changes to the SG geometry. These changes involved increasing the diameter of manways from 516mm to 600mm, thereby improving access and reducing exposure times. These improvements were expected to contribute dose savings of 2% to each of the collective doses associated with two of the substantial tasks listed in Section 4.8.1.4.1 above, namely inspections on reactor coolant side and repairs on reactor coolant side.

307 Thirdly, there were changes in the maintenance programme. Maintenance plans had been prepared for reactor coolant pressure boundaries, secondary system pressure boundaries and tube bundles. Improvements in these plans had benefits for radiological protection and these are summarised below.

- In the French fleet of NPPs, secondary system compartment welds (approx. half the areas to be inspected) were inspected on all four SGs during an outage. For the UK EPR, inspections were planned for only one SG during an outage (although results of inspections may indicate that further inspections would be required). This should reduce expected doses associated with SG ultrasonic testing by 50%.
- Eddy current tube inspections for the UK EPR would not be carried out during ROOs (which reflects current changes in the French fleet).
- Closed circuit television inspection of the tubesheet on the secondary system side would not be carried out during ROOs, but as the feasibility of this was still pending operational feedback data from the French fleet, no account was taken regarding benefits in terms of dose reductions.
- Lancing operations to extract sludge from the tubesheet would not be carried out during ROOs (since operational feedback data indicated that the amount of sludge extracted from SGs in the French fleet was sufficiently low to leave two cycles between lancing operations).

308 Fourthly, there was the option of channel head electropolishing. This process would provide a long-lasting reduction of contamination in the SG channel head, which would result in significant reductions in dose rates for SG operations on the reactor coolant side. Indeed, this should reduce collective expected doses by 25% for work activities associated with preparation, inspection and repairs on the reactor coolant side, SG bimetallic welds liquid penetrant testing and SG connection radiographic testing.

309 With regard to changes in the maintenance programme discussed above, it should be noted that in-service inspection (structural integrity) is outside the scope of GDA (Ref. 129). Therefore, it would be necessary for a future licensee to provide an ALARP justification (regarding structural integrity) for carrying out SG ultrasonic testing of secondary system component welds on only one SG during an outage (rather than on all four SGs), and for not carrying out eddy current tube inspections during ROOs. It follows, therefore, that it would also be necessary for that licensee to provide an ALARP justification (regarding radiological protection) to demonstrate worker dose optimisation for SG ultrasonic testing of secondary system compartment welds if more than one SG was inspected during an outage and if eddy current tube inspections were carried out during ROOs. I have captured this requirement in a GDA assessment finding, **AF-UKEPR-RP-09**, below (see Section 4.8.1.4.3).

4.8.1.4.1.2 Progress of Feasibility Studies for SG Inspection and Maintenance

310 EDF and AREVA were carrying out a number of feasibility studies designed to reduce exposures of workers, and these studies were ongoing (Ref. 85). At the time of writing the document, some of these studies had identified clear benefits in terms of dose reduction, some were still ongoing, whilst others had identified that there would be no or minimal benefits in dose reduction. The feasibility studies were as follows.

- Proposals for the supply of nozzle dams with inflatable seals: these had expected dose savings of approx. 5% for preparation work on the reactor coolant side.
- Making the primary manway bolt tensioning device more reliable, lighter and with improved performance levels: these had expected dose savings of approx. 10% for preparation work on the reactor coolant side.

- Optimising the insulation design by partial disassembly of eye and hand holes: although dose savings had been identified, they had been included in the optimisation study on fitting and removing insulation [see Sections 4.8.1.4.4 to 4.8.1.4.6 of this report) rather than in this optimisation study on SG inspection and tests.
- Design and implementation of removable decks for open eye and hand hole operations and adapted biological protection: the purpose of the removable deck was to fill the free space between the floor and the steam generator once the insulation had been removed. Dose savings were expected for preparation work on the secondary side system, but any savings were considered to be small, and so were not considered in the calculation of the collective dose to maintain conservatism.
- Reducing the height between the ground and the channel head: the purpose of this was to reduce access time during channel box operations, but after consideration by EDF and AREVA, it became clear that such height reductions would not be feasible and so there would be no impact on worker doses.
- Optimising leak recovery pipeline route to avoid hot spots: EDF and AREVA reviewed the pipelines that crossed through the steam generator bunkers since operational feedback data showed that these had a large impact on ambient dose rates in the reactor coolant and secondary systems. After consideration by EDF and AREVA, it became clear that there was little that could be done to reduce ambient dose rates, although the introduction of pipelines incorporating gravity-based draining towards sumps would at least limit the presence of hot spots. For the purposes of the study, therefore, no dose savings were assumed.
- Facilitating transport of tools on a trolley: the possibility of installing tracks was considered but dismissed on the grounds of building cleanliness and potential decontamination challenges. Although a number of provisions were put in place to facilitate the movement of trolleys, dose savings were not considered sufficient to be taken into account in collective dose calculations.
- Taking account of service requirements for maintenance activities: improvements to a number of systems were investigated with a view to reducing worker exposures. These included systems facilitating the installation of equipment hatches and handling hatches and tools (lifting lugs, support hooks, etc.), air drying systems, flow restrictors on steam generator channel head equipment hatches etc., and installation of lancing networks. These improvements would produce dose savings of 2% to the collective dose across the entire SG preparation and tests work activities. In addition, the installation of lancing networks should reduce expected doses associated with lancing by 5%.

4.8.1.4.1.3 Examples of SG Inspection and Maintenance

311 I raised a TQ on high dose work activities which included, amongst other things, optimisation of radiation doses for jumpers, details of work activities, management of exposures in timeframes measured in seconds, operational experience of dose control, and alternative means of carrying out work activities, including optimisation of doses using remote handling, remote observation and robotics (TQ-EPR-596, Ref. 8). However, since the selection of operation / maintenance practices and robotics would be the decision of future operators, towards the end of my assessment it was acknowledged that operation / maintenance practices and robotics was outside the scope of GDA.

Nevertheless, I have summarised some of the information provided regarding operation / maintenance practices and robotics to demonstrate opportunities for dose savings.

312 The response to TQ-EPR-596 (Ref. 8) stated that the work activities carried out by jumpers was work on SG channel heads for NDT, SG tube plugging, and visual inspections. EDF and AREVA provided a breakdown of radiation doses for manual and automatic tube plugging (response to TQ-EPR-596, Ref. 8), and NDT and visual inspection (response to TQ-EPR-1093, Ref. 8). Although operation / maintenance practices (which may include work activities carried out by jumpers) was outside the scope of GDA, the data provided was operational feedback data which provided information to demonstrate how doses associated with particular tasks may be broken down into its constituent parts.

4.8.1.4.1.4 Manual Tube Plugging and Testing

313 Manual tube plugging involved tube marking, tube plugging and plug testing (operational feedback data in response to TQ-EPR-596, Ref. 8). These tasks were made up of phases of work (e.g. worksite installation), with each phase comprising of a number of elementary tasks (e.g. annular space installation, hot leg temporary shielding, cold leg temporary shielding). Tube marking, tube plugging and tube testing each had several phases of work with each comprising a number of elementary tasks. Automatic tube plugging was similarly described in terms of phases and elementary tasks (response to TQ-EPR-596, Ref. 8).

314 Each elementary task was broken down into a number of elements, including the time spent by each individual in performing the elementary task (ranging from a few seconds to approx. an hour), the number of workers involved (usually an individual, but, some tasks required up to four workers), the dose rate in the location where the elementary task would be undertaken, and the integrated dose (in person-mSv). In this example, the temporary shielding installed for NDT had stayed in place.

315 There were approx. 50 elementary tasks associated with manual tube marking, plugging and testing. About half of these were activities with a duration of less than 30 seconds. Some of these short elementary tasks had a duration of a few seconds and were carried out in locations with dose rates around $50mSvh^{-1}$. The remaining tasks were carried out in locations with dose rates between a few $mSvh^{-1}$ and approx. $50mSvh^{-1}$.

316 The other elementary tasks were activities which lasted up to approx. an hour. About half of these longer elementary tasks were of relatively short duration (up to 16 minutes) and were carried out in locations with dose rates of less than $2mSvh^{-1}$. The remaining tasks lasted up to approx. an hour and were carried out in locations with dose rates between background and a few hundreds of $microSvh^{-1}$.

317 The integrated doses for the elementary tasks were not recorded for all those elementary tasks, but those that were recorded ranged from a worker carrying out an elementary task with a duration of approx. 20 minutes in a location with a negligible dose rate to a worker carrying out an elementary task with a duration of less than 30 seconds in a location with a dose rate of approx. $50mSvh^{-1}$. I briefly considered the integrated doses for the elementary tasks where integrated doses were not recorded and concluded that they would all be within this range.

318 It is difficult to estimate dose uptake for elementary tasks with durations of a few seconds since such estimates would be associated with large errors. For example, if a task was estimated to take a few seconds but a worker slipped or dropped a tool, it is reasonable

that the dose received could easily be two or three times larger than the estimated dose. But having said that, the estimated dose, though larger, might still be relatively small in magnitude.

319 This emphasises the benefits of breaking down tasks into elementary tasks to identify, for example, those elementary tasks with larger doses, or those where deviations / mishaps might easily result in larger doses, in order to identify areas which may achieve greatest benefits in dose reductions from ALARP improvements.

4.8.1.4.1.5 Automatic Tube Plugging and Testing

320 Automatic tube plugging and testing was undertaken by robotics, but still had several phases of work comprising a number of elementary tasks for workers to carry out (operational feedback data in response to TQ-EPR-596, Ref. 8). As for manual tube plugging, each elementary task for automatic tube plugging was broken down into a number of elements, including the time spent by each individual in performing the elementary task (ranging from a few seconds to approx. 40 minutes), the number of workers involved (usually an individual, but some tasks required up to three workers), the dose rate in the location where the elementary task would be undertaken, and the integrated dose (in person-mSv). Again, in this example, the temporary shielding installed for the NDT had stayed in place.

321 There were approx. 50 elementary tasks associated with automatic tube plugging and nearly half of these were activities with a duration of less than a minute. A few of these elementary tasks had a duration of a few seconds and were carried out in locations with dose rates around $50mSvh^{-1}$. Most were carried out in locations with dose rates from a few $microSvh^{-1}$ to less than $20mSvh^{-1}$.

322 The remaining elementary tasks were activities with a duration of up to approx. 40 minutes. About half of these elementary tasks had a duration of up to 5 minutes or less, and were carried out in locations with dose rates between a few $microSvh^{-1}$ and approx. $50mSvh^{-1}$. The remaining tasks were carried out in locations with dose rates of less than $500microSvh^{-1}$.

323 The integrated doses for the elementary tasks were recorded for all of the tasks, and ranged from a negligible integrated dose (for a worker carrying out an elementary task with a duration of a few minutes in a location with a dose rate of less than $5microSvh^{-1}$) to an integrated dose of less than 0.5person-mSv (for a worker carrying out an elementary task with a duration of approx. a minute in a location with a dose rate of less than $20mSvh^{-1}$.

324 As with the manual tube plugging, it is difficult to estimate dose uptake for elementary tasks with durations of a few seconds since such estimates would be associated with large errors, but the benefits of breaking down work activities into such elementary tasks can help to identify those tasks that may achieve greatest benefits in dose reductions from ALARP improvements.

4.8.1.4.1.6 Manual versus Automatic Tube Plugging and Testing

325 The integrated dose for automatic plugging and testing of two tubes was less than 2.5person-mSv which gave a reduction in dose of approx. 10% when compared with the integrated dose for the manual plugging and testing of two tubes. Even though robotics are outside the scope of GDA, this example illustrates that there are opportunities for

dose reductions using robotics for work activities such as tube plugging and testing, although dose savings need to be balanced against the time taken to set up the robotics and the number of tubes being plugged and tested.

4.8.1.4.1.7 NDT

326 The technique used for NDT of SGs was eddy current testing, and this testing was performed by robots (response to TQ-EPR-1093, Ref. 8). EDF and AREVA contacted three suppliers of robots with regard to actions required by workers in operating the robots, and there was a range of responses. For example, some suppliers required workers to enter the SG channel head to set up the robot, whereas others did not.

327 As for manual and automatic tube plugging and testing, EDF and AREVA provided operational feedback data for NDT that was provided for information to demonstrate how doses associated with particular tasks may be broken down into its constituent parts. The operational feedback data covered three work scenarios. These were NDT requiring access into the SG channel head without temporary shielding, access into the SG channel head with temporary shielding, and no access into the SG channel head without temporary shielding.

328 The work activities associated with NDT requiring access into the SG channel head involved several work phases made up from a number of elementary tasks. When temporary shielding was used there were additional elementary tasks for the installation and removal of temporary shielding. The elementary tasks were usually undertaken by an individual, but some tasks required a few workers.

329 When NDT requiring access into the SG channel was undertaken in the absence of temporary shielding, the elementary tasks ranged from a task with a duration of a few seconds in a location with a dose rate of approx. 40mSvh^{-1} to a task with a duration of several hours in a location with a dose rate of less than 5microSvh^{-1}. The integrated doses for elemental tasks were less than 5person-mSv.

330 When NDT requiring access into the SG channel was undertaken with temporary shielding, the elementary tasks ranged from a task with a duration of a few seconds in a location with a dose rate of approx. 40mSvh^{-1} to a task with a duration of several hours in a location with a dose rate of less than 5microSvh^{-1}, which were the same as when the work was carried out in the absence of shielding. The integrated doses for elemental tasks ranged from a few tens of person-microSv to less than 0.25person-mSv. Doses associated with some elemental tasks were unchanged, some were reduced by approx. 50%, and there were additional doses associated with the installation and removal of the temporary shielding. Overall there was a reduction in the total integrated dose with an overall dose saving of approx. 30% from using temporary shielding.

331 When NDT was undertaken that did not require access into the SG, no temporary shielding was considered necessary. The work involved several phases made up from a number of elementary tasks. This time the elementary tasks ranged from a task with a duration of a few seconds in a location with a dose rate of approx. 40mSvh^{-1} to a task with a duration of approx. 50 hours in a location with a dose rate of less than 5microSvh^{-1}. The integrated doses for elemental tasks were reported, and their total integrated dose was not dissimilar to the total integrated dose for NDT when access was required into the SG but where temporary shielding was used. There was an overall dose saving of approx. 20% compared with the total integrated dose for NDT when access was required into the SG but the work was carried out in the absence of temporary shielding.

332 As with the tube plugging and testing, it is difficult to estimate dose uptake for NDT for elementary tasks with durations of a few seconds, but the benefits of breaking down work activities into elementary tasks can help to identify those tasks that may achieve greatest benefits in dose reductions from ALARP improvements. In addition, even though robotics are outside the scope of GDA, this example again illustrates that there are opportunities for dose reductions using robotics for work activities such as NDT, although factors such as the time taken to set up robots and the use of temporary shielding may impact on dose savings.

4.8.1.4.1.8 Visual Inspection

333 The response to TQ-EPR-1094 (Ref. 8) explained that similar operational feedback data was not available for visual inspections since these were usually performed by the local SG maintenance team, and dose uptakes from these teams were registered on a general task code and individual doses were not available to EDF and AREVA design teams. In view of the amount of evidence already provided on sampling tube plugging and testing and NDT, I considered that this was sufficient for my assessment.

4.8.1.4.1.9 Use of Nozzle Dams

334 EDF and AREVA's proposal is that SG maintenance in the UK EPR will be undertaken when there is no fuel in the reactor vessel. Nevertheless, it is technically feasible to carry out SG maintenance with fuel in the reactor vessel, which might be desirable for unplanned SG maintenance without core unloading and loading, or if SG maintenance during an outage took longer than expected and the future operator wanted to begin preparations for start-up (response to TQ-EPR-987, Ref. 8). In such cases, SG nozzle dams would need to be installed to protect jumpers and / or robots from flooding. Nozzle dams were used in the French fleet of NPPs, and their installation had been optimised from a radiological protection point of view since they were designed to be fast mounting (a few minutes) and could be tightened at a distance (Ref. 51). However, carrying out SG maintenance with fuel in the reactor is outside the scope of GDA since it is an operational matter which would be for future operators to justify (see Section 2.3.6 above). In view of this, no safety case for nozzle dams was submitted or assessed during GDA.

4.8.1.4.2 Expectations – SG Inspections and Maintenance

335 The criteria / approach for developing the annual collective dose estimate for the UK EPR (see Section 4.8.1.1 above) are the same the criteria / approach for developing the collective dose estimates for SG maintenance and testing (see Section 4.8.1.4.1). As outlined in Section 4.8.1.1.5 above, guidance on strategies to restrict radiation exposure is in para. 479 of the SAPs (Ref. 4). This includes establishing a hierarchy of control measures, as discussed in paras 478 and 480 of the SAPs (Ref. 4), which in turn refer to guidance in L121 (Ref. 21) regarding IRR99 (Ref. 17). Guidance on strategies to restrict exposure is also in paras 4.1 and 4.2 of the TAG on Radiological Protection (Ref. 33), and guidance on the hierarchy of control measures is also in para. 4.3 of that TAG. In addition, guidance on remotely operated devices, which includes robotics, is in para. 4.1 of Ref. 33.

4.8.1.4.3 Findings – SG Inspections and Maintenance

336 In my opinion, from the evidence provided, EDF and AREVA had identified UK EPR improvements which would benefit radiological protection in terms of dose savings during SG inspections and maintenance (see Section 4.8.1.4.1.1 above), and feasibility projects which might drive down doses even further (see Section 4.8.1.4.1.2 above), This systematic review carried out by EDF and AREVA was appropriate and adequate. In particular, EDF and AREVA had identified UK EPR improvements that reduced doses received by workers carrying out SG inspection and maintenance activities that were dependent on the frequency of ultrasonic testing of secondary system compartment welds and eddy current tube inspections during outages (see Section 4.8.1.4.1.1). However, since in-service inspection (structural integrity) is outside the scope of GDA (Ref. 129), I have identified GDA assessment findings regarding in-service inspection of secondary system compartment welds and worker dose optimisation during ultrasonic testing of these welds and eddy current tube inspections as follows.

> ***AF-UKEPR-RP-09:*** *The licensee shall provide an ALARP justification (regarding structural integrity) for carrying out ultrasonic testing of secondary system component welds on only one SG during an outage (rather than on all four SGs), and for not carrying out eddy current tube inspections during ROOs. This shall be complete during the operational phase.*
>
> ***AF-UKEPR-RP-10:*** *The licensee shall provide an ALARP justification (regarding radiological protection) to demonstrate worker dose optimisation for SG ultrasonic testing of secondary system compartment welds if more than one SG is inspected during an outage, and for SG eddy current tube inspections if they are carried out during ROOs. This shall be complete during the operational phase.*

337 In my opinion, and in the opinion of the TSC, from the evidence provided, the final dose estimate for SG inspections and maintenance was acceptable. The final dose estimate for SG inspections and maintenance was conservative and assumed that no robotics were used. Although this final dose estimate of 22.6person-mSvy^{-1} (averaged over 10 years, and taking account of fewer inspections proposed for the UK EPR) was approx. four times higher than for operational feedback data from German KONVOI PWRs (Ref. 51), this was not entirely unexpected since dose estimates are invariably based on conservative calculations and are therefore conservative when compared with subsequent operational doses for those same work activities (Ref. 51). Therefore, this final dose estimate was considered acceptable.

338 In my opinion, and in the opinion of the TSC, from the evidence provided, the dose estimate for SG inspections and maintenance was acceptable. The final dose estimate for SG inspections and maintenance was conservative and assumed that no robotics were used. Although this final dose estimate of 22.6person-mSvy^{-1} was approx. four times higher than for operational feedback data from German KONVOI PWRs (Ref. 51), this was not entirely unexpected since dose estimates are invariably conservative compared with subsequent operational doses for those same work activities (Ref. 51). Therefore, this dose estimate was considered acceptable.

339 In the opinion of the TSC (Ref. 51), from the evidence provided, the breakdown of SG work activities into phases and elemental tasks provided for the UK EPR was suitable and sufficient, and was backed up by information in the ISOE database (Ref. 97).

340 In my opinion and in the opinion of the TSC, from the evidence provided, EDF and AREVA had adequately shown that there were opportunities for dose reductions using

robotics for SG maintenance and testing work activities such as tube plugging and testing and NDT (see Sections 4.8.1.4.1.6 and 4.8.1.4.1.7). I have identified a GDA assessment finding regarding the application of robotics in SG maintenance and testing as follows.

> ***AF-UKEPR-RP-11:*** *The licensee shall provide an ALARP justification for the use (or not) of robotics in SG maintenance and testing based on optimisation studies that identify specific tasks that should be carried out by specific robots. These tasks and robots shall be identified following a review of robots' capabilities for undertaking tasks that yield quantifiable benefits in terms of dose reductions for workers. This shall be complete during the operational phase.*

4.8.1.4.4 Assessment – Fitting and Removing Insulation

341 As stated previously, fitting and removing insulation was one of the seven work activities that contributed to 50% of the collective dose of the UK EPR (based on operational feedback data) (see section 4.8.1.1.4). EDF and AREVA had made a number of improvements that would reduce doses to workers carrying out these high dose work activities (Ref. 89). However, in contrast to the quantitative reviews and studies carried out by EDF and AREVA to identify improvements in terms of dose reductions for work activities associated with SG preparations and tests, the review for improvements in dose reductions associated with the fitting and removal of insulation was qualitative in nature.

342 My assessment considered the factors that would contribute to optimising doses during fitting and removing insulation, and the detailed design features that would enable those factors to be realised as dose savings during operation of the NPP.

343 The annual collective reference dose for fitting and removing insulation in the UK EPR was 30.6person-mSvy^{-1} (based on operational feedback data) (Ref. 89). The annual collective dose for one year was the average over a period of ten years, during which there would be a pattern of outages that included three NROs, two ROOs and one ISIO, as described in Section 4.8.1.1.4 above. During outages, insulation was removed to enable workers to gain access to plant and equipment so they could carry out work activities such as maintenance and testing, and then the insulation needed to be replaced when those work activities had been completed. The representative dose based on operational feedback data (30.6person-mSvy^{-1}) represented between 5% and 7% of the total annual collective for NROs and ROOs, and 13% for ISIOs. As for SG maintenance and testing, the reference dose for fitting and removing insulation was then reduced sequentially by taking credit for improvements designed specifically for the UK EPR. For improvements that had been identified but not yet implemented, the estimated annual collective dose fell to 17.9person-mSvy^{-1}, and for improvements that were still being studied the estimated annual collective dose was predicted to fall even further to 15.8person-mSvy^{-1}, which if achieved would be a dose saving of approx. 50%.

344 The purpose of this part of my assessment was to consider the evidence to support these predictions. Ref. 89 outlined 10 factors / requirements that would contribute to optimising doses and which could be implemented in the UK EPR without the need for detailed design studies. These 10 factors / requirements were as follows.

- Quick assembly / disassembly of heat insulation for the entire reactor coolant system, up to the second isolation component and throughout the entire main secondary system.
- Quick and independent assembly / disassembly of heat insulation and lagging on steam generators.

- Quick and independent assembly / disassembly of heat insulation on the pressuriser.
- Quick assembly / disassembly of insulation on welds and sensitive branch connections.
- Insulation location (corresponding to savings in dose of 10%).
- Additional lighting, electrical sockets and air inlets.
- Organised storage of insulation in predefined locations (not exposed to radiation if possible).
- Rest area (savings in dose of around 10% for some operations such as on hot loop, crossover leg, etc.).
- Use of adapted tools.
- Good practices applied to organisation: predefined schedule, organisation / preparation of operations.

345 Ref. 89 described the detailed design features that would enable these factors / requirements to be realised as dose savings during operation of the NPP. These detailed design features are summarised below.

- Verification of choice of heat insulation and lagging for the EPR: the use of heat insulation designed for quick assembly / disassembly was generally preferred for the UK EPR, but other factors also had to be considered, such as the advantages and disadvantages for insulating technologies, installation ergonomics and accessibility, and safety requirements.
- Comparison of insulation technologies with the French fleet of NPPs: the differences between the heat insulation of the FA3 EPR reactor coolant system and that used in the French fleet were assessed to demonstrate qualitative improvements in dose reductions for large equipment of the reactor coolant system. In the current fleet, insulation cassettes (which can be quickly assembled and disassembled) were only used for reactor coolant pipes, reactor coolant pump sets and the removable parts of the SG and pressuriser, and the remaining part of the isolated volume of the SG and pressuriser (more than 50%), used blanket insulation. In contrast to this, the FA3 EPR was entirely insulated using cassettes with the exception of two relatively small areas (on the expansion line and on the bottom of the pressuriser) which had to be insulated using blanket insulation.
- Implementation difficulties and solutions: in some cases, insulation assembly / disassembly could not be rapid due to the arrangement of the room. For example, in the safety injection system rooms (known as banana rooms), there were space-related challenges for fitting and removing insulation in terms of highly cramped areas and the position of equipment which required workers to bend over equipment to insulate it. In Section 3.3 of Ref. 89 there was a recommendation that specific insulation solutions should be provided for cramped areas (e.g. banana rooms, bottom of the pressuriser).
- Partial insulation removal: operational feedback data from the French fleet of NPPs showed that partial removal of insulation could represent significant savings in dose in cases where complete removal of insulation was not necessary for the work activity to be undertaken (e.g. for surge / spray lines, where dose savings of 85% may be achieved). Appendix B of Ref. 89 listed nine pieces of equipment where the insulation was most often removed and replaced (based on operational feedback

data). In Section 3.4 of Ref. 89 there was a recommendation that this list should be completed to identify all potential parts of the plant where partial insulation removal may assist in dose reduction. The UK EPR took account of non-quantifiable savings in dose from partial insulation removal.

- Weld inspections: all welds were insulated with removable cassettes that could be fitted and removed quickly (in addition to bolted covers, manholes and eye, hand and lancing holes), thereby reducing worker doses. Ref. 89 noted that doses received during ultrasound inspections of welds may be greater than during radiographic inspections since workers were in direct contact with the active piping during the ultrasound inspections. When the document was written, this part of the project was still ongoing so no impacts on dose could be indentified, and dose increases could not be ruled out.
- Electrical sockets, air inlets and additional lighting: the purpose of this part of the project was to identify the services required in the parts of the plant where workers received the greatest doses during maintenance work. These were interventions in close proximity to the refuelling machine, reactor vessel head inspection operations, operations on the steam generators, safety injection system / residual heat removal system exchanger inspection operations, operations on reactor coolant pumps, operations on the pressuriser, and maintenance of reactor coolant system, chemical volume and control system and safety injection system / residual heat removal system valves. Appropriate numbers of air inlets and electrical cabinets (of varying voltages) were installed as near as possible to the equipment requiring maintenance work, thereby making the work activity more efficient in terms of safety and time, and by reducing exposure times there were also dose savings. EDF and AREVA identified lighting specifications and standards, but since it was not known whether rooms were badly lit, the need for additional lighting could not be identified at this stage. The UK EPR took account of non-quantifiable savings in dose from improvements in the number and location of air inlets and electrical sockets.
- Use of adapted tools: EDF and AREVA identified a number of adapted tools that could be used and which may reduce doses during work activities. Since the selection of adapted tools would be the responsibility of future operators, dose savings could only be quantified during operations. Assessment of adapted tools is outside the scope of GDA, but the types of adapted tools identified by EDF and AREVA are listed here for information and completeness. Operational feedback data indicated that the use of front lamps on hard hats could reduce doses in poorly lit workstations by approx. 20%. Clear and accurate identification of insulation could help to reduce exposure times (this part of the study to identify dose savings was still ongoing). Scaffolding could be fitted with shielding by using lead screens to create safer rest areas to protect workers when they needed to wait for a few moments during their work activities, thereby reducing doses. Appropriate keys would be used to lock pieces of insulation together which would reduce the time required to disassemble that insulation. Ref. 89 also identified the importance of workers wearing appropriate personal and respiratory protective equipment to reduce doses.
- Packing areas: storing heat insulation and lagging (when it had been removed for maintenance and testing work activities) in predefined areas may contribute to optimisation of doses for fitting and removing insulation. A feasibility study was underway to identify packing areas where insulation material may be stored. This study took account of a number of factors, including installation ergonomics, storage in low dose rate areas, maintaining access through corridors, and potential

contamination and waste zoning. Packing areas were identified in a number of locations to store metal insulation material and foldable baskets to store blanket insulation material. Ref. 89 identified examples of potential storage areas which took account of preventing potential contamination by placing the containers as near as possible to the area where in insulation operation would be performed, placing insulation in wraps that could be decontaminated easily, and likewise providing containers with covers that could be decontaminated easily. Although a number of packing areas were identified, these areas would not be sufficient to store all the insulation parts. This emphasised the importance of performing partial insulation dismantling whenever possible (as discussed earlier in this Section). Although packing areas had been identified, there was insufficient data to draw definite conclusion with regard to dose optimisation, and given the lack of space for insulation storage, there may even be dose increases associated with packing areas.

- Rest areas: these were required to enable operators on standby to stay near to the workstation and for supervisors to monitor operations in low dose areas. Two possibilities were identified, namely rest areas associated with scaffolding (as discussed earlier in this Section) and rest areas primarily along the annuls for maintenance activities. Optimisation of doses would be studied later, but even at this stage EDF and AREVA were confident that there would be non-quantifiable dose savings.
- Best organisational practice: operational feedback data enabled a best practice guide to be developed for removing and fitting insulation in the main reactor coolant system during reactor coolant hydraulic tests. The results of the study were summarised in Ref. 89, which concluded that the use of best practice guides could contribute to reducing doses. Although good practice guides are not design features and are therefore outside the scope of GDA, this example is included to help to demonstrate the breadth of scope of the dose optimisation studies that EDF and AREVA has undertaken.

4.8.1.4.5 Expectations – Fitting and Removing Insulation

346 The criteria / approach for developing the annual collective dose estimate for fitting and removing insulation in the UK EPR (see Section 4.8.1.1 above) are the same the criteria / approach for developing the collective dose estimates for fitting and removing insulation (see Section 4.8.1.4.4). As outlined in Section 4.8.1.1.5 above, guidance on strategies to restrict radiation exposure is in para. 479 of the SAPs (Ref. 4). This includes establishing a hierarchy of control measures, as discussed in paras 478 and 480 of the SAPs (Ref. 4), which in turn refer to guidance in L121 (Ref. 21) regarding IRR99 (Ref. 17). Guidance on strategies to restrict exposure is also in paras 4.1 and 4.2 of the TAG on Radiological Protection (Ref. 33), and guidance on the hierarchy of control measures is also in para. 4.3 of that TAG. In addition, guidance on components used in high radiation areas being designed to be easily removable if maintenance is required is in para. 4.7 of Ref. 33.

4.8.1.4.6 Findings – Fitting and Removing Insulation

347 In my opinion and in the opinion of the TSC, from the evidence provided, EDF and AREVA had undertaken a systematic review of work activities which had identified improvements which may reduce doses during fitting and removing insulation. This systematic review carried out by EDF and AREVA, although still ongoing, was broad in its application and was appropriate and adequate.

348 In my opinion and in the opinion of the TSC, the final dose estimate for fitting and removing insulation of 15.8person-mSvy^{-1} (approx. 50% of the reference dose of 30.6person-mSvy^{-1}) appeared reasonable in view of the breadth of the systematic review. This review had considered 10 factors / requirements that would contribute to optimising doses, and had identified 10 detailed design features to enable these factors / requirements to be realised. Although some parts of the review were still ongoing, and some parts might indeed lead to increases in dose, overall there would be considerable dose savings even though these were not as yet quantifiable. There was no reason to suppose that the final dose estimate would not be achieved.

349 I have identified GDA assessment findings regarding fitting and removing insulation as follows.

> ***AF-UKEPR-RP-12:*** *The licensee shall provide an ALARP justification for fitting and removing insulation in cramped areas, and in particular, for fitting insulation in the safety injection system rooms (known as banana rooms) and at the bottom of the pressuriser. Any additional cramped areas where fitting insulation is challenging shall be identified following a review of cramped areas and their insulation requirements, and in cases where fitting insulation is challenging, those areas shall also be included in the safety case. This shall be completed during the operational phase.*
>
> ***AF-UKEPR-RP-13:*** *The licensee shall provide an ALARP justification for fitting and removing insulation where partial insulation removal is required for inspection and maintenance. The locations where partial insulation removal is required shall be identified following a review of work activities where complete removal of insulation would not be necessary for those work activities to take place, and of pieces of equipment where the insulation would be most often removed and replaced. This shall be completed during the operational phase.*

4.8.2 Expectations – Summary for Optimisation for Work Activities (Including Fuel Route)

350 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 17 and 21) and ND's TAG on the demonstration of ALARP (Ref. 32).

351 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. 34). 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. 34).

352 Para. 4.22 of Ref. 34 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. 34 explains that future operators would be required to have adequate arrangements for assessing the average dose to specific classes of persons.

353 Guidance relevant to optimisation for work activities (including the fuel route) on collective doses and high dose work activities, in particular, SG inspections and maintenance and fitting and removing insulation, is available as international guidance on the design of new NPPs published by the IAEA (Ref. 42) and NEA (Ref. 43), in the ISOE database of 2008 (Ref. 95), in IRR99 (Ref. 17) and its ACOP and guidance (Ref. 21), in ND's SAPs (Ref. 4), and in ND's TAG on Radiological Protection (Ref. 33).

4.8.3 Findings – Summary for Optimisation for Work Activities (Including Fuel Route)

354 In my opinion and in the opinion of the TSC, from the evidence provided, EDF and AREVA had provided sufficient information to show that the basis for the collective dose estimate was adequate, the strategy used to calculate the collective dose estimate was suitable and sufficient, the calculated average annual collective dose was acceptable and the future operational annual collective dose was expected to be lower than this value.

355 In my opinion, from the evidence provided, EDF and AREVA had identified UK EPR improvements which would benefit radiological protection in terms of dose savings during SG inspections and maintenance, and the systematic review carried out by EDF and AREVA was appropriate and adequate. In my opinion and in the opinion of the TSC, the final dose estimate for SG inspections and maintenance was based on conservative calculations, and since future operational doses for such work activities are expected to be lower, this dose estimate was considered acceptable. In the opinion of the TSC, from the evidence provided, the breakdown of SG work activities into phases and elemental tasks provided for the UK EPR was suitable and sufficient. Nevertheless, in my opinion and in the opinion of the TSC, from the evidence provided, EDF and AREVA had adequately shown that there were opportunities for further dose reductions using robotics for SG maintenance and testing work activities such as tube plugging and testing and NDT.

356 In my opinion and in the opinion of the TSC, from the evidence provided, EDF and AREVA had undertaken a systematic review of work activities which had identified improvements which may reduce doses during fitting and removing insulation. This systematic review carried out by EDF and AREVA, although still ongoing, was broad in its application and was appropriate and adequate. The final dose estimate for fitting and removing insulation appeared reasonable in view of the breadth of the systematic review. Although some parts of the review were still ongoing, and some parts might indeed lead to increases in dose, overall there would be considerable dose savings even though these were not as yet quantifiable. There was no reason to suppose that the final dose estimate would not be achieved.

357 There were three GDA assessment findings on robotics, fitting and removing insulation in cramped areas, and partial insulation removal. These were **AF-UKEPR-RP-11**, and **AF-UKEPR-RP-12** and **AF-UKEPR-RP-13**, respectively (see Sections 4.8.1.4.3 and 4.8.1.4.6 above).

4.9 Normal Operation – Waste Handling and Decommissioning

4.9.1 Assessment – Waste Handling and Decommissioning

358 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.

359 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 UK EPR Division 6 Assessment Report (Ref. 56). My assessment did not cover the adequacy of the arrangements for radioactive waste management and decommissioning, but rather I sampled the radiological protection aspects of radioactive waste and decommissioning. However, since the handling of radioactive waste was one of the seven high dose work activities that represented 50% of the total collective dose during normal operation (see Section 4.8.1.1.4 above), I decided to target my assessment 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.

4.9.1.1 Dose Optimisation for Handling Radioactive Waste

360 Operational feedback data showed that waste handling would likely represent 4.3% of the total annual reference collective dose of the UK EPR, where the annual collective dose for waste handling amongst the best units in the French fleet of NPPs was 19person-mSvy^{-1} (Ref. 88). In contrast to the other high dose work activities (which collectively, with waste handling, represented 50% of the annual collective dose), the majority of the dose received during waste handling was when the plant was operational rather than during shutdown, being approx. 20% of the total dose received during operation at power. However, this was because waste generated during a shutdown was stored temporarily and then processed during the next operational phase. My assessment sampled radiological protection challenges associated with solid waste treatment.

361 Ref. 88 showed that there were 11 activities that contributed to the dose associated with waste handling. These activities are listed below, along with the percentage of the total waste handling dose provided by EDF and AREVA as being associated with each work activity averaged over a period of three years. This information was operational feedback data. The type of shutdown during the three-year period was not specified, but this was not important since doses received during outages were relatively small compared with doses received during normal operation at power.

- Hulls (concrete waste container) plugging (3.3%).
- Encapsulation series with active resins (12.0%).
- Removal of radioactive waste (7.2%).
- Management of waste treatment (13.0%).
- Management of sludge treatment (11.7%).
- Management of treatment of steam generator blow down system resins (1.1%).
- Management of water filter treatment (11.7%).
- Management of checking of outside standard hulls (0.2%).
- Management of campaign active resins hulls (2.7%).
- Treatment work for other waste (15.3%).

- Sorting the waste in the Nuclear Auxiliary Building during shutdown (21.8%).

362 Sorting waste in the Nuclear Auxiliary Building during shutdown accounted for over 20% of the waste handling dose, and treatment of waste, including managing water filter treatment, accounted for 40% of the dose. Therefore, EDF and AREVA prioritised these work activities for dose optimisation.

363 EDF and AREVA had undertaken a review of waste handling and identified the following principles for optimisation of doses during sorting and treatment of waste.

- Mechanise and limit waste handling in order to reduce the effort and time involved and thereby reduce doses.
- Limit transfers and movements of the waste and splitting of the load.
- Treat and transport waste quickly to minimise temporary storage.
- Accelerate the treatment of technological waste produced in bags which represented the largest volume of waste with the reactor at shutdown. Technological waste included consumables (e.g. rags, cardboard boxes, latex gloves, plastics, oils, solvents), metallic waste, and specific waste (e.g. greasy or wet rags, aerosol sprays, glass, batteries, asbestos).
- Anticipate situations to prevent the accumulation of waste in areas not designed for waste storage, thus limiting the risk of fire and potential increase in ambient dose rates that waste may generate.
- Ensure that distinct, identified paths were available for treating the waste that were dependent on its nature, and that such paths were simplified to prevent errors in processing and in choice of the correct waste stream.

364 Ref. 88 described the waste processing activity in stages, identified improvements in the UK EPR that would reduce doses compared with the current French fleet of NPPs for particular processing activities, and indicated dose savings arising from those improvements.

365 Nuclear waste was divided into two groups, namely, technological waste (consumables, metallic waste, specific waste) and process waste (e.g. filters, resins, concentrates, sludges). For both types of waste, the waste processing chain involved several steps, namely, production and sorting, monitoring, treatment, storage, and removal. Processing of solid waste was further sub-divided into two streams, namely, low activity waste (with a surface dose rate less than $2mSvh^{-1}$) and medium activity waste (with a surface dose rate greater than $2mSvh^{-1}$). In addition to these activity criteria, wastes were also sorted at source according to whether or not they were compactable and the type of waste (standard waste, liquid waste, or special waste). More effort had been spent on mechanising the management of higher activity wastes than of lower activity wastes to prioritise worker dose optimisation.

4.9.1.2 Improvements for Processing Compactable Low Activity Waste

366 Medium activity waste represented the largest volume of waste that was generated, and it could risk overwhelming the processing chain at the end of a shutdown (Ref. 88). Improvements in the UK EPR design that may lead to dose reductions for radioactive waste handling compared with the current French fleet of NPPs are listed below.

- A specific room would be dedicated for the collection of waste from the nuclear island. This room would be used during operation at power and during outages, and was essentially a transit area en route to the sorting and drumming area of the Effluent Treatment Building. Operators would hand over bags of waste directly to "waste operators", and "waste operators" could advise on selective sorting thereby reducing errors. The reduced errors in sorting waste negated the need to repeat radiation monitoring checks at a later stage before compacting, and provided incentives to sort waste correctly at source.
- The layout of the compacting press room was improved in that it only dealt with the amount of waste required for the workstation nearby (the storage buffer room was elsewhere). The press was linked to a depressurising device to reduce the risk of contamination of workers. Bags were lowered by gravity between levels +7.40m and +3.70m, which limited the handling of bags and movement of skips (although waste could still be lowered in skips using the goods lift.
- The layout of the radiological inspection room was improved in that drums of compacted waste were transferred to a low dose rate area for weighing, labelling and radiological checking. Waste could then be directly loaded into containers ready for shipping which avoided multi-handling operations and thereby reduced doses.
- The layout of the waste storage room was improved in that it was designated as a clean room where only finished packages were stored. The containers were placed here ready to transport waste to the processing centre.

367 These improvements resulted in reductions in exposure times of 10%, reductions in ambient dose rates of 10%, and a reduction in the source term of 15%, all of which yielded dose savings.

4.9.1.3 Improvements for Processing Non-compactable Low Activity Waste

368 Non-compactable or special low activity waste was transported by an operator from the site to the shared waste collection room in the controlled area situated at level +7.40m in the Nuclear Auxiliary Building where it was sorted and placed in selective collection skips (Ref. 88). The low activity waste was then transported to the waste drumming room situated at the +7.4m level in the Effluent Treatment Building to be conditioned as appropriate for the required waste stream. Improvements in the UK EPR design that may lead to dose reductions compared with the current French fleet of NPPs are listed below.

- There was a single collection area for all the waste from the nuclear island so that wastes would not go astray.
- The "waste operators" were present to receive the waste material directly from operators, and the "waste operators" could advise on selective sorting thereby reducing errors.
- Selective sorting was carried out to avoid the need for repeat sorting before waste drumming thereby reducing waste handling resulting in dose savings.
- Waste could be crushed which reduced the volume of waste produced and limited the volume taken up in the Effluent Treatment Building, and the crushed waste receptacle was situated below the crusher. This simplified the introduction of waste at level +7.40m and permitted receipt of the waste without handling into a drum at level +3.70m for possible transfer to the compacting press, thereby reducing waste handling resulting in dose savings.

369 These improvements resulted in reductions in exposure times of 10%, reductions in ambient dose rates of 10%, and a reduction in the source term of 15%, all of which yielded dose savings.

4.9.1.4 Sorting, Checking, Storage and Removal of Medium Grade Technical Waste

370 Operational feedback data for one particular French NPP was studied to determine whether the arrangements for managing medium activity technical waste at that site would be suitable for the UK EPR (Ref. 88). Medium activity waste was placed in a shielded and padlocked rolling armoured skip positioned as close as possible to the maintenance site. The trolley had to be sufficiently manoeuvrable and small enough to be brought as close as possible to the working sites. This practice permitted the dose rate and transfer time to be limited up to the solid waste treatment system installation. When the site no longer generated this type of waste due to the NPP's position on the 10-year period of outage cycles, the skip could be moved to another site for use.

371 On closer examination, EDF and AREVA realised that some UK EPR installation constraints made this proposal unsuitable for the UK EPR (e.g. the presence of ground rails in the filter cartridge descent room which would make it difficult to manoeuvre the armoured trolley safely), although this may be revisited at a later stage. Therefore, no dose savings were attributed to use of the rolling armoured skip. Dose savings for these types of activities resulting from the reduction in the source term of 15% were taken into account.

4.9.1.5 Encapsulation of Process Waste

372 EDF and AREVA explained that there was a new proposed design for the Effluent Treatment Building which would be placed alongside the Nuclear Auxiliary Building and which shared a filter cartridge descent tube (Ref. 88). The filter removal machine (of the type used on the German KONVOI PWRs) was shared and was capable of removing filters from the Effluent Treatment Building and the Nuclear Auxiliary Building.

373 There were a number of advantages in this design which also improved radiological protection, and these are summarised below.

- The design was easy to operate since a single technology and operating mode was required for filter replacement, handling was reduced to a minimum, all filters were collected together in a single location, and there was only a single set of equipment to maintain which led to reductions in maintenance requirements and thereby dose savings.
- The design was simplified in that all the filters were brought together at the same level. The rearrangement of the rooms permitted the positioning of pumps, valves, filters and tanks in the same area which reduced the portions of the primary circuit that were irradiated. Also the trolley to transfer hulls between the Nuclear Auxiliary Building and the Effluent Treatment Building was no longer required.

374 The improvements in the UK EPR design that may lead to dose reductions compared with the current French fleet of NPPs are listed below.

- There was a reduced number of unencapsulated hull transfers outside the controlled area which reduced radiological risks and spread of contamination.
- Handling activities were reduced and simplified which resulted in dose savings.

- The installation permitted filter encapsulation so that only encapsulated waste would be in the storage room.
- A buffer area at the outlet of the locking cell was for the storage of packages waiting for plugging. This also allowed drying of the packages. Shielding would be provided to reduce radiation exposure of workers in the buffer area.
- The storage room was on level -3.90m and was made from concrete which helped to reduce dose rates at the exterior of the building.

375 These improvements are likely to result in reductions in exposure times and ambient dose rates and thereby reductions in doses. For water filters, reductions in doses were estimated at 20%, and reductions in doses for other processed wastes were estimated at 15%.

4.9.1.6 Overall Dose Savings for Radioactive Waste Handling

376 Ref. 88 explained that it was difficult to precisely quantify dose savings for waste handling because there was not a precise cut-off point at the end of each phase of the operational cycle for work activities associated with waste handling. Nevertheless EDF and AREVA estimated that doses received during waste handling for the UK EPR were 25% less than for the current French fleet of NPPs, which resulted in an estimated annual collective dose saving of 4.8person-mSvy^{-1} (from the annual reference collective dose of 19person-mSvy^{-1}.

4.9.2 Expectations – Waste Handling and Decommissioning

377 ND's TSC, AMEC, reviewed events involving radioactive waste and spent fuel recorded in IAEA's IRS, which showed that over 8100 incidents / events had been entered onto the system since 1980 (Ref. 55). Taking account of duplications and multiple reporting of the same events, the database contained 3579 extant entries relating to 551 nuclear reactors in 36 countries. Of these 3579 events, 273 related to radioactive waste and spent fuel. Worker exposure resulted from 32 of the incidents, and of these, only one had a design deficiency as being a root cause. This provided a useful benchmark.

378 There are ND TAGs on the management of radioactive materials and radioactive waste on nuclear licensed sites (Ref. 114), and on decommissioning on nuclear licensed sites (Ref. 115).

379 Guidance relevant to radioactive waste and decommissioning is available in some of the Radiological Protection Principles in the SAPs (Ref. 4). These are RP.3 on designated areas, RP.4 on contaminated areas and RP.5 on decontamination (plus the paragraphs of guidance associated with those principles), although the relevance is implicit rather than explicit.

380 The NEA guidance published in 2010 (Ref. 43) includes guidance on radiological protection principles and criteria for radioactive waste as they apply to designing new NPPs (Ref. 43).

4.9.3 Findings – Waste Handling and Decommissioning

381 In my opinion, from the evidence provided, the approach taken by EDF and AREVA to radioactive waste handling, including during decommissioning, was suitable and sufficient in that EDF and AREVA had undertaken a thorough review of waste handling to try to

identify improvements that might result in dose savings. These improvements resulted in reductions in exposure times, ambient dose rates and source terms, all of which yielded dose savings. There was no reason to suppose that the estimated annual collective dose saving of 25% (compared with the current French fleet of NPPs) would not be achieved.

382 There were no GDA assessment findings associated with waste handling and decommissioning regarding radiological protection.

4.10 Normal Operation – Public Exposure

4.10.1 Assessment – Public Exposure

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

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

384 As explained in Section 2.2 above, the regulation of public radiation exposure during normal operation is shared between EA and HSE, where IRR99 (Ref. 17) is enforced by ND on behalf of HSE, and EPR10 (Ref. 19) is enforced by EA. IRR99 (Ref. 17) 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. 21) 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, HPA RPD has recently recommended that doses to members of the public from new NPPs should be constrained to 0.15 mSv per year (Ref. 26).

385 Dose constraints for members of the public during normal operation were presented in Chapter 12 of the PCSR (Ref. 11). Dose estimates for members of the public during normal operation were presented in Chapter 11 of the PCER (Ref. 83). The estimated doses for local residents living 500m from the site were as follows.

- Total doses for an adult, child and infant were 25.8micro-Svy^{-1}, 11.6micro-Svy^{-1} and 11.0micro-Svy^{-1}, respectively. These doses consisted of aerial and liquid discharges plus direct radiation originating from within the site boundary.
- Doses from discharges for an adult, child and infant were 21micro-Svy^{-1}, 9.1micro-Svy^{-1} and 9.3 micro-Svy^{-1}, respectively.
- Doses from direct radiation originating from within the site boundary were 4.8micro-Svy^{-1}, 2.5 micro-Svy^{-1} and 1.7micro-Svy^{-1}, respectively.

386 EA assessed the doses to members of the public from aerial and liquid discharges during normal operation (Ref. 117), and ND assessed the doses to members of the public from direct radiation originating from within the site boundary during normal operation.

387 The percentage of the total dose to a member of the public from direct radiation originating from within the site boundary was small compared with the doses from aerial and liquid discharges, being 19%, 22% and 15% of the total dose for an adult, child and infant, respectively.

388 Direct radiation originating from within the site boundary was attributable to the bulk shielding plus shielding provided by other structures within the NPP. The topic of shielding was assessed by ND's TSC, NT, and the TSC's assessment that showed that bulk shielding provisions for a UK EPR unit were adequate to protect members of the public living 100m from such a unit from direct radiation during normal operations (see Section 4.3.1.7.1 above).

389 As described in Section 2.3 above, public radiation exposure during accident conditions is covered in the Step 4 PSA assessment report (Ref. 46) with regard to plume dispersion modelling and dose consequences. There are emergency reference levels and countermeasures which should be implemented to avert doses to people off the site (Refs 39 and 40).

4.10.2 Expectations– Public Exposure

390 BSLs and BSOs for any person off the site in normal operation are covered in NT-1 Target 3 of the SAPs and in paras 585 and 590 to 593 (Ref. 4). General guidance on radiological analysis in normal operation (the majority of which is relevant to persons off the site) is provided in paras 4.1 to 4.13 of T/AST/043 (Ref. 34), More specific guidance on radiological analysis for persons off the site (NT.1 Target 3) is in paras 4.24 to 4.28 (Ref. 34).

391 Legislation and guidance relevant to public exposure from direct radiation originating from within the site boundary is provided in Section 4.3.1.7.2 above. The pubic dose limit for members of the public (referred to as "other persons" in IRR99 (Ref. 17)) is the same as the BSL in the SAPs (Ref. 4), namely $1mSvy^{-1}$. Legislation on protection of members of the public from ionising radiation is IRR99 (Ref. 17), REPPIR (Ref. 18) and EPR (Ref.19). Guidance on protection of members of the public from ionising radiation is in the ACOP and guidance to IRR99 (Ref. 21), in guidance to REPPIR (Ref. 22), in NT.1 Target 3 in the SAPs (Ref. 4), and in guidance on new NPPs from HPA RPD (Ref. 26). Guidance on radiation shielding is in T/AST/002 (Ref. 29). This legislation and guidance is discussed in Section 2.2 above.

4.10.3 Findings – Public Exposure

392 As reported in Section 4.3.1.7.3 above, in the opinion of the TSC, the bulk shielding provisions for the UK EPR to protection members of the public were adequate. I concur with the opinion of the TSC from the evidence provided on protection of the public from direct radiation.

393 There were no GDA assessment findings associated with public exposure during normal operation.

4.11 Accident Conditions – Persons On-site During Accident Conditions

4.11.1 Assessment – Persons On-site During Accident Conditions

394 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.

395 Minor incidents arising from normal operation (e.g. reactor start-up, power operation, shutdown, maintenance, testing refuelling) which might give rise to operational problems or small unplanned doses to workers should be regarded as part of normal operation (not fault conditions) (Ref. 34). Impacts to people off-site during accidents are covered by Level 3 PSA, which is reported in the Step 4 PSA assessment report for the UK EPR (Ref. 46).

4.11.1.1 Assessment - Criticality Accidents

396 My assessment of criticality safety / accidents in the spent fuel pool is covered in Appendix A.

397 To assist in my assessment on criticality safety / accidents, I raised TQ-EPR-595 (Ref. 8) that was broad in context to provide an overview of information and a framework from which to undertake the specialist assessment. I also raised TQ-EPR-1018 (Ref. 8) that requested information / documentation on boron concentration in criticality control, and TQ-EPR-1019 (Ref. 8) that requested information / documentation on criticality control of fuel in dry and wet storage. I was supported in my assessment by ND's TSC, GRS. Guidance on criticality control is in ND's engineering principles on criticality safety, ECR.1 and ECR.2, and in paras 470 to 475 of the SAPs (Ref. 4). Further advice is available in ND's TAGs on criticality safety (Ref. 44) and criticality warning systems (Ref. 45).

398 The scope, expectations and findings of my assessment are provided in Appendix A to this report. In summary, criticality assessment covered the design and operational features intended to prevent the onset of an uncontrolled neutron chain reaction. This criticality assessment covered the handling of fuel (whether irradiated or not) from receipt and storage of fresh fuel to transfer of fuel within the fuel pond and subsequent storage. EDF and AREVA had indicated that in the long term, spent fuel may be stored either in another on-site pond or in dry storage on the site. Neither of these options was considered in any detail in this assessment but there is enough experience in the UK to give confidence that a pond could be designed to accommodate the output from the sixty year operating life of an EPR. Experience in dry fuel storage will continue to accumulate pending the licensees' decisions on the preferred methodology.

399 My assessment covered new fuel racks and the fuel pond. The fuel handling and storage systems had been designed with due regard to criticality safety. Systems had been modelled with conservative assumptions with regard to manufacturing tolerances and degradation.

400 To conclude, I am satisfied with the claims, arguments and evidence laid down within the Consolidated GDA Safety Submissions for criticality safety. I consider that from a criticality safety view point, the EDF and AREVA UK EPR design is suitable for construction in the UK. There are no GDA Issues arising from this assessment.

401 I have identified three GDA assessment findings regarding criticality control, regarding, the assurance of the presence of borated stainless steel in accordance with the design intent, monitoring borated stainless steel over the lifetime of the plant, and systems to

control and verify the enrichment and continued presence of boron used in the fuel pond (see Appendix A and Annex 1). These are repeated below.

> ***AF-UKEPR-RP-18*** *(see Appendix A): The licensee shall take steps at the construction stage to assure the presence of borated stainless steel in the fuel pond storage racks in accordance with the design intent. This shall be complete before fuel on-site.*
>
> ***AF-UKEPR-RP-19*** *(see Appendix A): The licensee shall establish systems to monitor the borated stainless steel in the fuel pond storage racks over the lifetime of the plant so as to identify and quantify any degradation. This shall be complete before fuel on-site.*
>
> ***AF-UKEPR-RP-20*** *(see Appendix A): The licensee shall establish systems to control and verify the enrichment of the boron used in the fuel pond and its continued presence in the fuel pond during its operation. This shall be complete before fuel on-site.*

4.11.1.2 Assessment – Non-criticality Accidents

402 To begin my assessment of non-criticality accidents arising on the nuclear island, I raised TQ-EPR-592 (Ref. 8) that requested information / documentation on doses to people on-site during potential accidents. This TQ was broad in context to provide an overview of information and a framework from which to undertake the specialist assessment. This TQ included the range of matters taken from Table 3 of my Step 4 Plan (Ref. 1) (with the exception of criticality accidents), such as exposure and evacuation times (see Section 4.11.1 above). I was supported in my assessment by ND's TSC, TUV SUD (Ref. 51).

4.11.1.2.1 Assessment – Impacts On-site of Design Basis Fault Sequences – NT.1 Target 4

403 NT.1 Target 4 in ND's 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. Design basis faults are associated with reactor-containment buildings.

404 My assessment considered radiological consequences of design basis events to people on the site. 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 UK EPR (Ref. 118). Although site-specific calculations for design basis radiological consequences are outside the scope of GDA, it was necessary to know that the radiological consequences predicted for a generic site could be compared favourably with established UK frequency targets for GDA.

405 A representative list of 13 design basis accidents were described in Sub-Chapter 14.6 of the PCSR (Ref. 11). The radiological consequences analysis was based on a mix of German and French requirements, and although these were not always consistent with ND's expectations, the ND assessor was satisfied that it should be possible for future site-specific analyses of design basis faults to show compliance with Target 4 of the SAPs. Therefore, an assessment finding was raised by the ND fault studies assessor that required future operators to provide site-specific radiological consequences analysis for design basis events (including hazards), taking due cognisance of usual UK methodology and assumptions and explicitly comparing the results against NT.1 Target 4 (see Annex 1 of Ref. 118). In view of this, it should also be possible for future operators

to undertake site-specific analyses of design basis faults whilst taking due regard to NT.1 Target 4 for people on the site.

4.11.1.2.2 Expectations – Impacts On-site of Design Basis Fault Sequences – NT.1 Target 4

406 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 (Ref. 4). Guidance on radiological analysis of fault conditions is provided in TAG T/AST/045 (Ref. 35) 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 (Ref. 4).

4.11.1.2.3 Findings - Impacts On-site of Design Basis Fault Sequences – NT.1 Target 4

407 Potential doses to workers on the site during accidents were not analysed in detail by EDF and AREVA and so were not compared with NT.1 Target 4 during GDA. In my opinion, this was not unreasonable at this stage in the design process because, apart from doses received by workers directly involved in the accident itself, EDF and AREVA reported that doses to workers on the site may generally be lower than those to people off the site since people on the site can be more easily and immediately protected by evacuation and sheltering in a safe area (Ref. 11). In my opinion, if the containment remained intact or if any release was via a stack at height, then doses on-site would be low, and this coupled with evacuation and shelter would reduce doses further. In addition, ND's TAG on radiological analysis under fault conditions advised that simple bounding estimates for on-site doses should normally be sufficient, particularly if the calculated doses are subject to relatively large uncertainties (Ref. 35); such bounding estimates were not provided.

408 I have identified a GDA assessment finding regarding impacts of design basis fault sequences / events to workers on the site as follows.

> ***AF-UKEPR-RP-14:*** *The licensee shall provide a safety case that demonstrates that the on site-specific radiological consequences analyses for design basis events (including hazards) are ALARP and have taken due cognisance of usual UK methodology assumptions and have explicitly compared the results of those analyses against NT.1 Target 4 in ND's SAPs regarding the predicted initiating fault frequency versus dose to individuals on the site. This shall be complete before fuel on-site.*

4.11.1.2.4 Assessment - Impacts On-site of Accidents – NT.1 Targets 5 and 6

409 NT.1 Targets 6 and 8 in ND's SAPs (Ref. 4) provide targets for a range of effective doses received by any person from accident scenarios (other than design basis faults) to people on the site and to members of the public off the site, respectively. Radiological consequences of such accidents to people off the site are considered in ND's Step 4 PSA Assessment Report (Ref. 46) regarding NT.1 Target 8 (which is linked with Target 7). My assessment considered radiological consequences of such accidents to people on the site regarding the risk impact to individuals from all facilities on the site (in Target 5) and the predicted single accident frequency versus dose to individuals (in Target 6). I did not consider severe accident analysis and societal risk regarding NT.1 Target 9.

410 Individual risks to members of the public from accident scenarios involving non-core damage sequences plus core damage sequences were shown in Sub-Chapter 17.4 of

the PCSR (Ref. 11). The cumulative frequencies for dose bands 1 (0.1–1mSv), 2 (1–10mSv), 3 (10-100mSv), 4 (100-1000mSv) and 5 (>1000mSv) in NT.1 Target 8 of the SAPs (Ref. 4) were $1.4x10^{-3}$, $1.3x10^{-5}$, $8.8x10^{-7}$, $6.5x10^{-8}$ and $5.6x10^{-8}$, respectively, in the PCSR (Ref. 11). I did not assess the methodologies used to calculate the cumulative frequencies for off-site doses.

411 These cumulative frequencies for individual risks to the public were lower than the BSOs in NT.1 Target 8. In particular, the cumulative frequencies for dose bands 1 and 5 were approximately one order of magnitude less that the BSOs, and dose bands 2, 3 and 4 were approximately two orders of magnitude less that the BSOs. Also, all the cumulative frequencies were an additional two orders of magnitude below the BSLs in NT.1 Target 8.

412 Individual risks to workers from accident scenarios were discussed in Sub-Chapter 17.3 of the PCSR (Ref. 11), which concluded that doses to workers on the site who were not in the vicinity of the accident would generally be lower than doses to people off the site for any particular accident scenario since people on the site could be more easily and immediately protected by evacuation and sheltering in a safe area. Although NT.1 Target 6 of the SAPs (Ref. 4) identified BSOs and BSLs for predicted frequencies per annum for four dose bands for any person on the site for any single accident, there was no equivalent table in the PCSR (Ref.11) that compared on-site effective dose bands against annual cumulative frequencies for people on the site as there were for people off the site in Sub-Chapter 17.4.

413 I raised TQ-EPR-1284 (Ref. 8) which requested information / documentation which compared the effective dose bands (in mSv) in NT.1 Target 6 of the SAPs (Ref. 4) against cumulative frequencies (per year) for individuals on the site during accidents (for example, similar to the information indicated above for individuals off the site during accidents). I also requested information / documentation on the methodologies and assumptions used to identify dose bands and cumulative frequencies for individuals on the site during accidents.

414 In its response to TQ-EPR-592, EDF and AREVA explained that it did not plan to address potential doses to workers on the site with the BSOs and BSLs in NT.1 Target 6 within the framework of GDA.

4.11.1.2.5 Expectations - Impacts On-site of Accidents – NT.1 Targets 5 and 6

415 Guidance on NT.1 Target 6 on accidents for any person on the site is in paras 602 to 606 of the SAPs (Ref. 4). Guidance on radiological analysis of fault conditions is provided in TAG T/AST/045 (Ref. 35) in paras 4.1 to 4.8, 4.19, and 4.22 to 4.23. Guidance on radiation protection during accident conditions is provided in RP.2 and paras 480 – 483 of the SAPs (Ref. 4).

4.11.1.2.6 Findings - Impacts On-site of Accidents – NT.1 Targets 5 and 6

416 Potential doses to workers on the site during accidents were not assessed in detail by EDF and AREVA and so were not compared with NT.1 Target 6 during GDA. In my opinion, this was not unreasonable at this stage in the design process because, as described in Section 4.11.1.2.3 above, apart from doses received by workers directly involved in the accident itself, EDF and AREVA reported that doses to workers on the site may generally be lower than those to people off the site since people on the site can be more easily and immediately protected by evacuation and sheltering in a safe area (Ref. 11). In addition, ND's TAG on radiological analysis under fault conditions advised that

simple bounding estimates for on-site doses should normally be sufficient, particularly if the calculated doses are subject to relatively large uncertainties (Ref. 35); such bounding estimates were not provided.

417 I have identified a GDA assessment finding regarding impacts of accidents to workers on the site as follows.

> ***AF-UKEPR-RP-15:*** *The licensee shall provide a safety case that demonstrates that the on site-specific radiological consequences analyses for accidents (including hazards) are ALARP and have taken due cognisance of usual UK methodology assumptions and have explicitly compared the results of those analyses against NT.1 Target 5 in ND's SAPs regarding the risk impact to individuals from all the facilities on the site, and against NT.1 Target 6 in ND's SAPs regarding the predicted single accident frequency versus dose to individuals on the site. This shall be complete before fuel on-site.*

4.11.1.2.7 Assessment - Escape Routes and Accessibility to Plant Immediately Post Accident

418 I raised TQ-EPR-1285 (Ref. 8) to request information on escape routes based on French national requirements that were being used for the FA3 EPR. I also requested clarification of any local operator actions required immediately post accident by workers on the affected plant, including the control room, for accidents with potential off-site radiological consequences.

419 The escape routes for workers as soon as an accident occurred were based on a fire code (ETC-F) that was sent to ND as part of the assessment of internal hazards. The response to TQ-EPR-1285 (Ref. 8) summarised these requirements and the key factors as follows.

- Escape routes depended on the number of persons likely to be in that part of the plant and any particular first aid requirements.
- There were specific requirements on the maximum number of steps per staircase, there had to be sufficient space for stretchers to be manoeuvred on half-landings, staircases going down towards the ground floor had to be separate from those going up towards the ground floor. and there was a maximum distance between the bottom / top of staircases on the ground floor and the outside of the building.
- Lifts or hoists could not be used to compensate for reductions in the number or width of escape routes.
- There were specific requirements on maximum distances that workstations could be from escape staircases / corridors, and on maximum lengths of dead ends.

420 I considered the adequacy of escape routes in terms of radiological protection (not in terms of fire safety). Ref. 69 showed that all corridors, staircases and emergency exits were in green radiological zones with ambient dose rates of less than 25microSvh^{-1}.

421 The response to TQ-EPR-1285 (Ref. 8) explained that operator actions required to manage the plant immediately post accident could be done from the main control room on safety classified equipment, so there was no need for workers to go to other parts of the plant to carry out any specific functions.

422 EDF and AREVA had undertaken a study to demonstrate the habitability of the main control room for a period of five days post accident (response to TQ-EPR-1285, Ref. 8).

If the ventilation system was operating after the accident, then the dose rate in the main control room would be less than 16microSvh^{-1}. If the ventilation system did not operate, then the dose rate would rise but should remain less than 1.2mSvh^{-1}, which EDF and AREVA considered should make the main control room accessible for a period of 12 hours immediately post accident, and which was sufficient time for the operator actions referred to above to be carried out to manage the plant. If a single operator remained in the control room for a period of 12 hours, then dose received would be approx. 14mSv, which would be less than the annual dose limit for workers in IRR99 (Ref. 17). If the main control room needed to be occupied for more than 12 hours immediately post accident, then future operators would need to provide an ALARP justification for occupancy for longer periods of time (see Section 4.11.1.2.9 below).

423 ND's fault studies assessor had requested ND's TSC, Serco, to review off-site radiological consequences from design basis accidents for the UK EPR (Ref. 118). The Serco report (Ref. 120) considered three of the thirteen representative design basis accidents that were described in Sub-Chapter 14.6 of the PCSR (Ref. 11). These three design basis accidents were SG tube rupture, fuel handling accidents and large break loss of coolant accidents. My intention was to review the on-site radiological consequences from two of these three design basis accidents, namely SG tube rupture and fuel handling accidents, since I and the TSC for radiological protection anticipated that these may have the greatest impact on doses to workers on the site. However, since the response to TQ-EPR-1285 (Ref. 8) explained that all accidents could be managed from the main control room and there was no need for workers to go to other parts of the plant to carry out any specific functions, then this area of my assessment was not followed through.

424 However, in contrast to this, in the particular case of design basis events caused by fire, or where fire was a contributing factor, local actions could be performed by operators to help with fire management. In such cases, all local actions could be performed from the following types of locations.

- Outside the radiological designated area.
- In green radiological zones with ambient dose rates of less than 25microSvh^{-1}.
- In pale yellow radiological zones with ambient dose rates of less than 0.2mSvh^{-1}.
- In deep yellow radiological zones with ambient dose rates of less than 2.0mSvh^{-1}.

425 For design basis events involving fire, the actions required outside the main control room in the pale and deep yellow radiological zones related to the closure of particular valves, and since the time required for such actions was limited, this meant that doses could be controlled during these valve closures. In all cases, the valve locations (in pale and deep yellow radiological zones) were accessible from green radiological zones. These arrangements were appropriate in terms of dose management.

426 ND's TSC report (Ref. 51) explained that in the last generation of German PWRs, most accident scenarios could be handled from outside the containment, but some specific scenarios required workers to access the containment. However, since the UK EPR was designed without the need to access inside the containment during accident conditions, and following discussions with ND's fault studies assessors, I did not pursue this matter further.

4.11.1.2.8 Expectations - Escape Routes and Accessibility to Plant Immediately Post Accident

427 Guidance relevant to access and escape routes for fire fighting and operational personnel are in WENRA reference levels referred to in Appendix 1 of ND's TAG on internal hazards (Ref. 119). Guidance emphasising the importance of evacuation times is in para. 4.8 of TAG T/AST/045 (Ref. 35). Guidance on radiation protection during accident conditions is provided in RP.2 and paras 480 – 483 of the SAPs (Ref. 4).

4.11.1.2.9 Findings - Escape Routes and Accessibility to Plant Immediately Post Accident

428 In my opinion and in the opinion of the TSC, from the evidence provided, the placement of escape routes through low radiation zoned areas was appropriate. Dose rates in the main control room would not be insignificant immediately post accident if the ventilation system failed, and occupancy over time periods greater than 12 hours should be justified. Accessing valves for closure (in locations with higher radiation zoned areas) via low radiation zoned areas was appropriate in terms of dose management. Although no dose uptake predictions were provided for workers on the site, the dose rate data provided in Ref. 69 and in the response to TQ-EPR-1285 (Ref. 8) provided sufficient evidence that procedures could be put in place by future operators to ensure that valves could be closed immediately post accident whilst doses to workers remained ALARP.

429 I have identified a GDA assessment finding regarding occupancy of the main control room immediately post accident as follows.

> ***AF-UKEPR-RP-16:*** *The licensee shall provide an ALARP justification for occupancy of the main control room immediately post accident if the ventilation system has failed. This shall be complete before fuel on-site.*

4.11.1.2.10 Assessment - Accessibility to Plant during Post-Accident Conditions

430 Sub-Chapter 12.5 of the PCSR (Ref. 11) defined the systems, rooms and components for which access would be required in post-accident conditions for long-term cooling of the plant and fuel pond (over a period of one year), and identified work that would be necessary before access could take place.

431 The response to TQ-EPR-1285 (Ref. 8) explained that accessibility of the rooms was required for up to one year after the accident occurred for maintenance and / or repair of components (pumps, valves and fans), and once the maintenance or repair had been carried out then the rooms would no longer be accessed.

432 Further studies on accessibility in post accident conditions were planned (Ref. 121), which identified the need to increase the list of access requirements in the post-accident phase in Sub-Chapter 12.5 of the PCSR (Ref. 11). These studies were performed by a working group comprised of engineers specialising in a range of areas, including equipment design, systems operation, radiation protection, source term and radiation protection calculations, and safety (Ref. 121). The working group identified the following work streams.

- Inventory of rooms and equipment requiring access for maintenance / repair activities (e.g. inventory of relevant rooms and systems, knowledge of the maintenance / repair requirements for equipment).
- Access conditions (e.g. dose rates in rooms where equipment requiring maintenance / repair was located and in access routes to those rooms, evaluation of airborne

contamination in the buildings, evaluation of temperature in the rooms and access routes.

- Access routes (e.g. planned access routes for maintenance / repair activities during normal operation, alternative access routes depending on ambient conditions (such as dose rates and temperature) and on tool dimensions).
- Inventory of and delays to maintenance / repair activities (e.g. knowledge of the maintenance programme for equipment during post-accident conditions which may be adapted from the maintenance programme for equipment during normal operations, specifications of interventions for each type of equipment based on operational feedback data (although UK EPR-specific equipment may not have such data), evaluation of accessibility conditions for each type of equipment whilst taking account of longer and more frequent operational delays).
- Preliminary actions to allow maintenance / repair activities to take place (e.g. draining or rinsing pipes to reduce dose rates in rooms where dose rates were too high to carry out maintenance / repair of equipment, and which might involve the design of new rinsing devices as a matter of priority).
- Isolation of equipment requiring maintenance / repair (e.g. the feasibility of isolation, arrangements for protection of workers during equipment isolation, equipment modifications to allow isolation of equipment that has been damaged / degraded).

433 The FA3 EPR was the reference design for the UK EPR, and so the post-accident accessibility study for the FA3 EPR should demonstrate post-accident accessibility requirements for the UK EPR. Ref. 121 stated that the studies discussed above were undertaken for the FA3 EPR. The report of the results of the study was due for submission during Step 4 through TQ-EPR-592 (Ref. 8). Unfortunately, there were delays in the preparation and submission of the report (Ref. 122) which was not available until my assessment had been completed. Therefore, in view of the breadth of the study undertaken, future operators should review accessibility to plant during post-accident conditions for the UK EPR and ensure that doses to workers accessing equipment under post-accident conditions are ALARP see Section 4.11.1.2.12 below.

4.11.1.2.11 Expectations - Accessibility to Plant during Post-Accident Conditions

434 In ND's TAG on radiological analysis of fault conditions, T/AST/045 (Ref. 35), para. 4.6 explains that doses associated with recovery actions by workers following faults should be excluded from on-site radiological analysis of NT.1 Targets 4 and 6. Instead, dose management for recovery actions by workers should be carried out under IRR99 (Ref. 17), and in particular under the hierarchy of control measures as described in para. 478 of the SAPs (Ref. 4). Guidance on radiation protection during accident conditions is provided in RP.2 and paras 480 – 483 of the SAPs (Ref. 4). Advice on IRR99 (Ref. 17) is in its ACOP and guidance in L121 (Ref. 21).

4.11.1.2.12 Findings - Accessibility to Plant during Post-Accident Conditions

435 In my opinion, from the evidence provided, and in view of the breadth of the study undertaken for accessibility to plant during post-accident conditions for the FA3 EPR, future operators should review accessibility to plant during post-accident conditions for the UK EPR and ensure that doses to workers accessing equipment under post-accident conditions are ALARP.

AF-UKEPR-RP-17: *The licensee shall provide a safety case to identify access requirements to specific components / pieces of equipment that will require maintenance / repair during the post-accident phase, and to identify potential doses to workers carrying out those maintenance / repair activities and to demonstrate that they are ALARP. This shall be complete before fuel on-site.*

4.11.2 Expectations – Summary for Persons On-site During Accident Conditions

436 Guidance on criticality control is in ND's engineering principles on criticality safety, ECR.1 and ECR.2, and in paras 470 to 475 of the SAPs (Ref. 4). Further advice is available in ND's TAGs on criticality safety (Ref. 44) and criticality warning systems (Ref. 45).

437 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 (Ref. 4). Guidance on NT.1 Target 6 on accidents for any person on the site is in paras 602 to 606 of the SAPs (Ref. 4). Guidance on radiological analysis of fault conditions is provided in TAG T/AST/045 (Ref. 35) in paras 4.1 to 4.8, 4.17 to 4.19, and 4.22 to 4.23. Guidance on radiation protection during accident conditions is provided in RP.2 and paras 480 – 483 of the SAPs (Ref. 4).

438 Guidance relevant to access and escape routes for fire fighting and operational personnel are in WENRA reference levels referred to in Appendix 1 of ND's TAG on internal hazards (Ref. 119). Guidance emphasising the importance of evacuation times is in para. 4.8 of TAG T/AST/045 (Ref. 35).

439 Legislation relevant to dose management for recovery actions by workers is in IRR99 (Ref. 17), and guidance on the hierarchy of control measures is described in para. 478 of the SAPs (Ref. 4). Advice on IRR99 (Ref. 17) is in its ACOP and guidance in L121 (Ref. 21).

4.11.3 Findings – Summary for Persons On-site During Accident Conditions

440 In my opinion, from the evidence provided, there should be three GDA assessment findings on future operators regarding criticality control, namely, on the assurance of the presence of borated stainless steel in accordance with the design intent, monitoring borated stainless steel over the lifetime of the plant, and systems to control and verify the enrichment and continued presence of boron used in the fuel pond (see Section 4.11.1.1 above, Appendix 1 and Annex 1).

441 Potential doses to workers on the site during accidents were not assessed in detail and so were not compared with NT.1 Targets 4 and 6 during GDA. In my opinion, from the evidence provided, this was not unreasonable at this stage in the design process.

442 In my opinion and in the opinion of the TSC, from the evidence provided, the placement of escape routes through low radiation zoned areas was appropriate. EDF and AREVA had provided sufficient evidence that procedures could be put in place by future operators to ensure that valves could be closed immediately post accident whilst doses to workers remained ALARP.

443 Although the UK EPR was designed without the need to access inside the containment during accident conditions, future operators may choose / have to carry out some specified actions inside the containment immediately post accident. Therefore, in my opinion and in the opinion of the TSC, future operators should re-evaluate the need for access to containment immediately post accident, and if necessary, ensure that doses to workers accessing containment under such conditions are ALARP.

444 In my opinion, from the evidence provided, and in view of the breadth of the study undertaken for accessibility to plant during post-accident conditions for the FA3 EPR, future operators should review accessibility to plant during post-accident conditions for the UK EPR and ensure that doses to workers accessing equipment under post-accident conditions are ALARP.

445 There were four GDA assessment findings on impacts of design basis fault sequences / events to workers on the site, impacts of accidents to workers on the site, occupancy of the main control room immediately post accident and review of accessibility to plant during post-accident conditions for the UK EPR. These assessment findings were **AF-UKEPR-RP-14, AF-UKEPR-RP-15, AF-UKEPR-RP-16 and AF-UKEPR-RP-17** (see Sections 4.11.1.2.3, 4.11.1.2.6, 4.11.1.2.9 and 4.11.1.2.12, respectively, above).

4.12 Accident conditions – Intervention Personnel

4.12.1 Assessment – Intervention Personnel

446 My Step 4 Plan (Ref. 1) explained that my assessment of intervention personnel would include the following matters.

- Facilities and design features for responding to accidents.
- Optimisation of doses to intervention personnel.
- Optimisation of doses for saving life.

447 Sub-Chapter 12.5 of the PCSR (Ref. 11) identified legislation relevant to emergency preparedness and intervention personnel, namely IRR99 (Ref. 17), REPPIR (Ref. 18) and the Nuclear Installations Act 1965, as amended (Ref. 20). This legislation was supplemented with guidance in L121 (Ref. 21), L126 (Ref. 22) and ND's SAPs (Ref. 4) / TAGs, respectively. Sub-Chapter 12.5 of the PCSR (Ref. 11) also identified upper dose levels for personnel involved in intervention and in saving life (Ref. 23). Emergency preparedness arrangements with regard to preparing emergency plans, identifying dose levels for intervention personnel etc., are the responsibility of future operators. Nevertheless, there needs to be sufficient space at the correct locations to set up control centres for managing emergency response at the tactical level on the site.

448 As part of its response to TQ-EPR-1285 (Ref. 8), EDF and AREVA outlined, for information only, the types of emergency facilities that were appropriate for the UK EPR. These were based on French requirements for the FA3 EPR, although they would also need to take account of both national legal requirements and future operator requirements in due course. The facilities included the following matters.

- Alarms (a speaker network and an alarm network to provide verbal information and warning signals from the main control room to workers).
- A telecommunications system dedicated to emergency situations (e.g. independent telephone networks).
- Data transfer system (e.g. fax, dedicated computer network).
- Emergency telephone directories and contact details.
- Emergency rooms for a number of functions (e.g. muster points outside the radiological designated area for workers and visitors to the site, safe areas set aside for workers and visitors to shelter with appropriate welfare arrangements, facilities to

support the distribution of stable iodine, areas for storage and distribution of equipment required during an emergency).

449 In addition to the above, many of which are outside the scope of GDA, it is also important to ensure suitable access points into the nuclear island to enable intervention personnel to enter the area to deal with an accident, and this is within the scope of GDA.

450 The dose limits in IRR99 do not apply during intervention (Ref. 18). Dose limitation of intervention personnel is achieved through identifying dose levels for emergency exposures which are higher than the dose limits in IRR99 (Ref. 17) in order to protect plant and prevent a release of radioactivity. Higher dose levels for emergency exposures may be authorised for the purposes of saving life (Refs 22 and 23). Even so, emergency exposures must be ALARP, and REPPIR require operators to notify HSE at the emergency planning stage of dose levels for emergency exposures for intervention personnel (Refs 22 and 23). As already discussed in Section 4.11.1.2.11 above, accessibility to plant to undertake repair / maintenance work during post-accident conditions would be undertaken under IRR99 (Ref. 17) and its ACOP and guidance (Ref. 21).

4.12.2 Expectations – Intervention Personnel

451 Legislation relevant to emergency preparedness and intervention personnel is IRR99 (Ref. 17), REPPIR (Ref. 18) and the Nuclear Installations Act 1965, as amended (Ref. 20). This legislation is supplemented with guidance in L121 (Ref. 21), L126 (Ref. 22) and ND's SAPs (Ref. 4) / TAGs, respectively.

4.12.3 Findings – Intervention Personnel

452 TQ-EPR-1453 (Ref. 8) requested information / documentation on the layout of areas set aside for accessing controlled areas, including information / documentation relevant to health physics facilities for entering and exiting controlled areas. Although the role of the health physics facilities in emergency response was not explicit in the TQ raised, nevertheless, such facilities need to be able to accommodate the additional workloads which may be put upon it, either directly or in concert with a nearby facility, during an accident. I raised Assessment Finding **AF-UKEPR-RP-02** on access to controlled areas which included the ability of the facilities to accommodate additional workloads which may be put upon them by intervention personnel during accident conditions (see Section 4.2.4 above).

453 Apart from the link to the layout of areas set aside for accessing controlled areas and possible locations for relevant health physics facilities for entering and exiting controlled areas as discussed above, there were no GDA assessment findings associated with intervention personnel.

4.13 Overseas Regulatory Interface

454 HSE's Strategy for working with overseas regulators is set out in Refs 123 and 124. In accordance with this strategy, HSE collaborates with overseas regulators, both bilaterally and multi-nationally.

4.13.1 Bilateral Collaboration

455 HSE's 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

4.13.2 Multilateral Collaboration

456 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.

457 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.

458 I also attended a meeting on radiological protection under the auspices of the NEA MDEP EPR Working Group. This meeting was hosted by NEA and attended by representatives from ND, Environment Agency, ASN, NRC and STUK. I attended the meeting with ND's radioactive waste and decommissioning assessor and with colleagues from the Environment Agency. Experiences were shared regarding radiological protection and radioactive waste aspects of the EPR.

459 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 to broadly similar standards.

4.14 Interface with Other Regulators

460 I have worked closely with the Environment Agency through the whole of GDA. Future operators of the UK EPR 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.

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

4.15 Other Health and Safety Legislation

462 In addition to the legislation identified in Section 2.2 above, 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, as amended (MHSWR99) (Ref. 125) and its ACOP and guidance (Ref. 126). 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. 17), whereas the general requirement to review and revise all occupational risk assessments is in MHSWR99 (Ref. 126).

5 CONCLUSIONS

463 This report presents the findings of the Step 4 Radiological Protection assessment of the EDF and AREVA UK EPR reactor.

464 The Step 4 assessment in my topic area commenced with consideration of the relevant chapters of the PCSR and supporting references available at that time, and these are referred to as appropriate in this report. As the GDA submission developed during Step 4, in response to my regulatory questions, amendments were made as appropriate to the PCSR (Ref. 11) and its supporting references (which are listed in the Submission Master List (Ref. 12)). A review has been made of the updates to the GDA submission in my technical topic area and the conclusion of this review is that the updates to, or information included in, the GDA submission and / or supporting references were not as expected and further work is required to address these shortfalls. This will be progressed in GDA through my GDA Issue **GI-UKEPR-RP-01**. In this technical topic area my assessment is therefore limited to the versions of the GDA submission documents referred to in my assessment report. Although the cPCSR (Ref. 13) and its supporting references (which are listed in the Submission Master List (Ref. 16)) are therefore acceptable as the reference point for an interim Design Acceptance Confirmation (iDAC) these outstanding issues require acceptable resolution before a final Design Acceptance Confirmation (DAC) can be issued.

465 To conclude, I am broadly satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for radiological protection. I consider that from a radiological protection view point, the EDF and AREVA UK EPR design is suitable for construction in the UK. However, this conclusion is subject to satisfactory progression and resolution of the GDA Issue 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

466 In my opinion, from the evidence provided, EDF and AREVA had designed the plant and its operations to ensure that engineered features would restrict exposures of workers to ionising radiation so far as is reasonably practicable during normal operation. This had been done by reducing source terms, ensuring bulk shielding was adequate (subject to the GDA issue on radiological zoning and bulk shielding), providing hot workshops for decontamination of plant and equipment, providing sample transfer systems, providing a two room concept in the Reactor Building to enable workers to enter the service space with the reactor at power, locating radiation monitoring equipment that required regular maintenance and testing in low dose rate areas, and providing strategies for decontamination of plant during operation, maintenance, POCO and decommissioning. The bulk shielding also provided adequate protection for members of the public from direct radiation originating within the site boundary.

467 In my opinion, from the evidence provided, EDF and AREVA had undertaken a suitable and sufficient systematic and comprehensive approach to optimising exposures of workers to ionising radiation when carrying out high dose work activities. This programme had involved experts across a range of disciplines. The programme had identified the work activities where workers received the highest doses, and had then taken a staged approach to identify ways to reduce these doses. To do this, EDF and AREVA had identified a reference dose for a particular work activity based on the best

performing PWRs in the French fleet of NPPs, had identified reductions in exposures from modifications to the work activities that could be implemented, and then had gone on to explore other possible means of reducing doses even further. This systematic and staged approach had resulted in predicted reductions in exposures to workers undertaking, for example, SG inspections and maintenance, fitting and removing insulation, and waste handling, compared to doses received in the best performing PWRs in the French fleet of NPPs.

468 In my opinion, from the evidence provided, EDF and AREVA had taken appropriate steps to design the plant and its operations to ensure that engineered features would restrict exposures of workers to ionising radiation so far as is reasonably practicable during accident conditions. This had been done by providing escape routes, providing protection to the main control room to enable workers to remain there for a minimum of 12 hours and possibly much longer immediately post accident, and providing means to operate components and equipment without the need to enter containment immediately post accident. EDF and AREVA had also identified requirements to provide safe access to components requiring maintenance and repair during a period of one year after an accident. EDF and AREVA had not assessed potential doses to workers during accidents in detail which was not unreasonable at this stage in the design process.

5.1.1 Assessment Findings

469 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

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

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Table 1

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
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

Table 1

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
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
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

Table 1

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 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 (SP) 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 – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-RP-01	**Source Terms:** The licensee shall provide procurement procedures that require a review of materials associated with the primary coolant before purchase of those materials from their supplier in order to identify if there are any improvements in reductions in levels of cobalt or any other elements in materials which might lead to further reductions in radiation exposure of workers, and which would not compromise the functionality of those materials.	This shall be complete before mechanical, electrical and control and instrumentation systems are delivered to site.
AF-UKEPR-RP-02	**Designated areas:** The licensee shall provide a report to demonstrate that the layout of the health physics facilities for entering and exiting the controlled area are suitable and sufficient, in particular, provision of rooms and equipment (including PPE / RPE) shall be arranged to minimise the potential spread of contamination. The report shall also include the ability of the facilities to accommodate additional workloads which may be put upon them by intervention personnel during accident conditions.	This shall be complete before first structural concrete.
AF-UKEPR-RP-03	**Shielding:** The licensee shall provide a report to identify areas where temporary shielding will be required for specific work activities and ensure there is adequate space for storage of such shielding when not in use, ideally in low dose rate areas near to the location where the work activities will be undertaken.	This shall be complete before fuel on-site.
AF-UKEPR-RP-04	**Ventilation:** The licensee shall provide an ALARP justification to establish the ventilation conditions in the equipment compartment and service space of the Reactor Building, including the airborne concentrations of radionuclides, under which workers may enter the service space with the reactor at power, in order to ensure that internal doses to workers are ALARP.	This shall be complete before fuel on-site.

Annex 1

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

Radiological Protection – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-RP-05	**PRMS:** The licensee shall provide a report to demonstrate that the control and instrumentation aspects of the installed radiation monitoring equipment of the KRC and KRT systems of the PRMS are adequate.	This shall be complete before mechanical, electrical and control and instrumentation safety systems inactive commissioning.
AF-UKEPR-RP-06	**PRMS:** The licensee shall provide a report to demonstrate that the planned location of the installed radiation monitoring equipment of the KRC and KRT systems of the PRMS are appropriate and take account of the final radiological zoning classification scheme with regard to ensuring that radiation exposures received by workers whilst taking measurements or maintaining or testing such equipment are ALARP	This shall be complete before mechanical, electrical and control and instrumentation safety systems inactive commissioning.
AF-UKEPR-RP-07	**Decontamination:** The licensee shall provide a report to demonstrate its site-specific strategies, systems and techniques for decontamination during operations and maintenance, and during POCO and decommissioning, whilst taking account of the contamination zoning of the NPP.	This shall be complete before fuel on-site.
AF-UKEPR-RP-08	**Decontamination:** The licensee shall provide a report to demonstrate its strategy for laundry provision, and if that strategy includes an on-site laundry, then that laundry shall be designed such that its location and containment integrity will ensure that doses to workers are ALARP.	This shall be complete before first structural concrete.
AF-UKEPR-RP-09	**Optimisation for work activities:** The licensee shall provide an ALARP justification (regarding structural integrity) for carrying out ultrasonic testing of secondary system component welds on only one SG during an outage (rather than on all four SGs), and for not carrying out eddy current tube inspections during ROOs.	This shall be complete during the operational phase.

Annex 1

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

Radiological Protection – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-RP-10	**Optimisation for work activities:** The licensee shall provide an ALARP justification (regarding radiological protection) to demonstrate worker dose optimisation for SG ultrasonic testing of secondary system compartment welds if more than one SG is inspected during an outage, and for SG eddy current tube inspections if they are carried out during ROOs.	This shall be complete during the operational phase.
AF-UKEPR-RP-11	**Optimisation for work activities:** The licensee shall provide an ALARP justification for the use (or not) of robotics in SG maintenance and testing based on optimisation studies that identify specific tasks that should be carried out by specific robots. These tasks and robots shall be identified following a review of robots' capabilities for undertaking tasks that yield quantifiable benefits in terms of dose reductions for workers.	This shall be complete during the operational phase.
AF-UKEPR-RP-12	**Optimisation for work activities:** The licensee shall provide an ALARP justification for fitting and removing insulation in cramped areas, and in particular, for fitting insulation in the safety injection system rooms (known as banana rooms) and at the bottom of the pressuriser. Any additional cramped areas where fitting insulation is challenging shall be identified following a review of cramped areas and their insulation requirements, and in cases where fitting insulation is challenging, those areas shall also be included in the safety case.	This shall be completed during the operational phase.
AF-UKEPR-RP-13	**Optimisation for work activities:** The licensee shall provide an ALARP justification for fitting and removing insulation where partial insulation removal is required for inspection and maintenance. The locations where partial insulation removal is required shall be identified following a review of work activities where complete removal of insulation would not be necessary for those work activities to take place, and of pieces of equipment where the insulation would be most often removed and replaced.	This shall be completed during the operational phase.

Annex 1

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

Radiological Protection – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-RP-14	**Persons on-site during accident conditions:** The licensee shall provide a safety case that demonstrates that the on site-specific radiological consequences analyses for design basis events (including hazards) are ALARP and have taken due cognisance of usual UK methodology assumptions and have explicitly compared the results of those analyses against NT.1 Target 4 in ND's SAPs regarding the predicted initiating fault frequency versus dose to individuals on the site.	This shall be complete before fuel on-site.
AF-UKEPR-RP-15	**Persons on-site during accident conditions:** The licensee shall provide a safety case that demonstrates that the on site-specific radiological consequences analyses for accidents (including hazards) are ALARP and have taken due cognisance of usual UK methodology assumptions and have explicitly compared the results of those analyses against NT.1 Target 5 in ND's SAPs regarding the risk impact to individuals from all the facilities on the site, and against NT.1 Target 6 in ND's SAPs regarding the predicted single accident frequency versus dose to individuals on the site.	This shall be complete before fuel on-site.
AF-UKEPR-RP-16	**Persons on-site during accident conditions:** The licensee shall provide an ALARP justification for occupancy of the main control room immediately post accident if the ventilation system has failed.	This shall be complete before fuel on-site.
AF-UKEPR-RP-17	**Persons on-site during accident conditions:** The licensee shall provide a safety case to identify access requirements to specific components / pieces of equipment that will require maintenance / repair during the post-accident phase, and to identify potential doses to workers carrying out those maintenance / repair activities and to demonstrate that they are ALARP.	This shall be complete before fuel on-site.
AF-UKEPR-RP-18	**Criticality control:** The licensee shall take steps at the construction stage to assure the presence of borated stainless steel in the fuel pond storage racks in accordance with the design intent.	This shall be complete before fuel on-site.

Annex 1

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

Radiological Protection – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-RP-19	**Criticality control:** The licensee shall establish systems to monitor the borated stainless steel in the fuel pond storage racks over the lifetime of the plant so as to identify and quantify any degradation.	This shall be complete before fuel on-site.
AF-UKEPR-RP-20	**Criticality control:** The licensee shall establish systems to control and verify the enrichment of the boron used in the fuel pond and its continued presence in the fuel pond during its operation.	This shall be complete before fuel on-site.

Note: It is the responsibility of the Licensees / Operators to have adequate arrangements to address the Assessment Findings. Future Licensees / Operators can adopt alternative means to those indicated in the findings which give an equivalent level of safety.

For Assessment Findings relevant to the operational phase of the reactor, the Licensees / Operators must adequately address the findings during the operational phase. For other Assessment Findings, it is the regulators' expectation that the findings are adequately addressed no later than the milestones indicated above.

GDA Issues – Radiological Protection – UK EPR

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT
GDA ISSUE
RADIOLOGICAL ZONING AND BULK SHIELDING
GI-UKEPR-RP-01 REVISION 0

Technical Area	**RADIATION PROTECTION**		
Related Technical Areas	Civil Engineering		
GDA Issue Reference	**GI-UKEPR-RP-01**	**GDA Issue Action Reference**	**GI-UKEPR-RP-01.A1**
GDA Issue	Radiological zoning for restriction of exposure to ionising radiation of workers is fundamental to the design of the nuclear island, and bulk shielding is inextricably linked with civil engineering aspects of that design. The radiological zoning classification scheme underpinned by design shielding calculations is not referenced in the GDA submission for the UK EPR design.		
GDA Issue Action	Provide an overview document that supplements the claims and arguments presented in the PCSR Chapter 12.3 with additional information on the radiological zoning classification scheme for the nuclear island, including dose rate criteria and predictions for all modes of plant operation, for occupied areas as a direct reference from the PCSR. A radiological zoning classification scheme should be provided to demonstrate that there is adequate shielding provision for all areas of the facility. This should be presented as an overview document that provides information / documentation which summarises the dose rates and radiological classifications within all rooms and for all modes of plant operation (for example, power operation, outages, refuelling). The document should include information / documentation on the Reactor Building, Fuel Building, Safeguard Building, and Auxiliary Building. The overview document should summarise the results of shielding calculations to show that the predicted dose rates within each area of the plant meet the radiological classification. The response should, as a minimum, summarise the following information for each room of the facility: • Room descriptor and number / designation. • Radiological classification (namely dose rate criteria). • Dose rate prediction(s) for each room giving the maximum dose rate present during all modes of operation (for example, power operation, outages, refuelling). • Reference to shielding assessments / calculations containing data regarding the assumed radiation sources, shielding provisions and calculated dose rates. With agreement from the Regulator this action may be completed by alternative means.		

Further explanatory / background information on the GDA Issues for this topic area can be found at:	
GI-UKEPR-RP-01 Revision 0	Ref. 138.

Appendix A

Criticality Control in the Spent Fuel Pool for the EDF and AREVA UK EPR™ Reactor

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1 INTRODUCTION

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

2 Completion of Step 4 represents the end of our planned GDA assessment on the topic of criticality safety for the EDF and AREVA UK EPR.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR CRITICALITY SAFETY

2.1 Assessment Plan

3 The Assessment Plan for Radiological Protection of the EDF and AREVA UK EPR (Ref. 4) identified the criticality safety of the stored fuel, in particular within the fuel pond as one of the areas to be assessed during Step 4 of GDA.

2.2 Standards and Criteria

4 The proposal has been compared against HSE's Safety Assessment Principles (SAP) (Ref. 4) which specifically cover criticality and the relevant engineering SAPs.

5 The SAPs set an expectation that the fuel storage design should use passive systems to ensure sub-criticality with a hierarchy based on engineering controls in preference to administrative systems 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.

6 HSE's SAPs (Ref. 4) 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 which have been specifically written to address spent fuel storage and criticality (Refs 131, 132). Although still in draft these guides provide useful support to the SAPs and an indication of up-to-date international experience in this area.

2.3 Assessment Scope

7 Criticality assessment covers the design and operational features intended 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 using the double contingency principle and, where necessary supported by a probabilistic assessment of the fault scenarios.

8 This criticality assessment covers the handling of fuel (whether irradiated or not) from receipt and storage of fresh fuel to transfer of fuel within the fuel pond and subsequent storage. It does not however, cover emplacement or arrangement of fuel within the core. These aspects fall within the scope of ND's Step 4 Assessment of Fuel and Core Design (Ref. 133).

9 EFD and AREVA have indicated that in the long term, spent fuel may be stored either in another on-site pond or in dry storage on the site. Neither of these options is considered in any detail in this assessment but there is enough experience in the UK to give confidence that a pond could be designed to accommodate the output from the sixty year operating life of an EPR. 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. 135) and I was able to see fuel being prepared for dry fuel storage at a PWR in the US. Experience in dry fuel storage will continue to accumulate pending the licensees' decisions on the preferred methodology.

10 EDF and AREVA's criticality safety case claims to show that the behaviour of MOX fuel is bounded by modelling fuel elements as unirradiated elements of 5w% ^{235}U. My assessment has not considered this claim and therefore the conclusions of this assessment cannot be assumed to be valid for MOX fuel.

2.3.1 Findings from GDA Step 3

11 ND's criticality assessment did not start until Step 4 so there are no findings from Step 3.

2.3.2 Use of Technical Support Contractors

12 GRS was engaged to perform a detailed examination of the content of the safety related submissions to assess whether the proposed criticality safety measures are suitable and sufficient with regard to the ALARP principle. The organisation:

- reviewed whether the control measures specified by EDF and AREVA 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;
- compared the safety case against SAPs and international guidance;
- to perform a detailed examination of the content of the safety related submissions to assess whether the proposed criticality safety measures are suitable and sufficient with regard to the ALARP principle; and
- run independent calculations (Ref. 53) using the Monte Carlo code KENO Va and found that EDF and AREVA's calculations were reproducible.

2.3.3 Cross-cutting Subjects

13 There are no cross-cutting topics directly relevant to fuel storage.

2.3.4 Integration with other Assessment Topics

14 In-core fuel handling is considered in the Step 4 Fault Studies assessment report (Ref. 118). Other aspects of the spent fuel pond are addressed in Step 4 assessment reports considering Fault Studies, Mechanical Engineering (Ref. 100) and Civil Engineering (Ref. 137).

2.3.5 Out of Scope Items

15 No items have been agreed with EDF and AREVA as being outside the scope of GDA on this topic.

3 REQUESTING PARTY'S SAFETY CASE

3.1 New Fuel racks

16 The dry fuel store comprises stainless steel racks with locations for twelve assemblies inside a concrete vault in the fuel building (Ref. 11). The assemblies are held on a pitch of 535mm giving a separation between cells of 289mm. No neutron poison is incorporated in the design.

17 The design prevents the insertion of an assembly between two adjacent cells. The vault is normally operated dry and no water sources are routed through the vault. But in considering flooding EDF and AREVA have surveyed the effects of water of various densities and found that optimum moderation is produced by full density water. Their calculations show that in the event of inadvertent flooding the distance between the assemblies afforded by the design ensures that the system remains sub-critical.

3.2 Spent Fuel Pond

18 The EDF and AREVA UK EPR fuel pond is intended to store new fuel awaiting loading into the reactor core and spent fuel assemblies until their shipment from the plant for further interim storage elsewhere or disposal, while removing the decay heat emitting from spent fuel (Ref. 11). The pond consists of a concrete structure in which racks are provided to maintain a defined separation between fuel elements. When the pressure vessel head is removed e.g. for refuelling, the fuel pond water and primary coolant are able to mix. The pond water will therefore be dosed with boric acid manufactured with boron with an enriched ^{10}B content – consistent with the boron specification for the primary circuit. This provides protection against inadvertent dilution of the boron content of the primary circuit.

19 EDF and AREVA have used the code MCNP4 to perform the deterministic analysis to allow the racks to be designed for fresh and irradiated fuel to meet a criterion for normal conditions of $K_{eff} < 0.95$ allowing for uncertainties. For accident conditions the comparable design criterion is $K_{eff} < 0.98$.

20 EDF and AREVA have put forward a case based on the use of racks constructed from stainless steel with 2mm thick borated stainless steel box sections of internal width 222mm within each cell to enhance neutron absorption. The 1167 locations thus formed comprise a grid of storage cells of pitch 275mm. The fuel assemblies are 214mm square so giving a nominal separation of 61mm. Primary criticality control is maintained by the geometrical spacing defined by the racking system together with the neutron absorption properties of the boron in the borated stainless steel.

21 Some of the locations along the edges of the racks are blocked off preventing nucleonic interaction with an element dropped between the pond wall and the racks. In addition five specific locations are provided with filters to cater for the possibility of damaged elements.

22 Further protection in accident conditions is provided by the boric acid dissolved in the pond water which is conservatively modelled as 2000ppm.

23 In order to derive a bounding model of a fuel assembly EDF and AREVA performed calculational surveys to examine the effect on reactivity of removing fuel pins. EDF and AREVA found that reactivity peaked when 3 fuel pins were omitted and this representation was used as a conservative model. Fuel assemblies were modelled with a number of other conservative approximations including:

- ^{235}U of enrichment 5.1w%;
- manufacturing tolerances are conservatively represented; and
- burnable poisons in the fuel elements have not been modelled.

24 The system is modelled flooded with unborated water and shown to meet the criterion of $K_{eff} < 0.98$ under otherwise normal conditions. Under the intended conditions of boron concentration in excess of 2000ppm EDF and AREVA show that K_{eff} comfortably meets the criterion of $K_{eff} <0.95$.

25 Accident scenarios such as dropped or misplaced elements are modelled with 2000ppm boron and the system is shown to meet the criterion of $K_{eff}<0.98$; a simultaneous loss of boron is regarded as a double contingency.

26 The EDF and AREVA UK EPR will use enriched boron but the deterministic calculations have been performed by modelling the boron content of the pond water as natural boron, which has a ^{10}B proportion of about twenty percent - the remainder being ^{11}B which has no significant neutron absorption properties in the thermal energy region. This assessment follows that convention by referring to equivalent concentrations of natural boron.

27 The boron content of the pond will be verified by sampling, the periodicity of which will be established as part of the operational case.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR CRITICALITY SAFETY

28 The arrangements and design criteria for fuel storage are described in Chapter 9.1 of Ref. 11 but TQ-EPR-595 (Ref. 8) was raised to obtain more detailed information of the calculational methodology.

4.1 New Fuel Racks

29 Under normal (i.e. dry conditions) the system is sub-critical by a large margin. EDF and AREVA have calculated that full density water produces optimum moderation, and when flooding with water is postulated the separation of 289mm ensures that neutronic interaction between elements is very limited. EDF AREVA have calculated a conservative value of K_{eff} of 0.93455 for the racks flooded with water, which is comfortably within the criterion of K_{eff}<0.98.

4.1.1 Assessment

30 The new fuel racks maintain a separation between the cells of 289mm. Experience shows that separation in water of 300mm provides effective neutronic isolation in low enriched uranium systems. Accordingly EDF and AREVA calculate that the value of K_{eff} under fully flooded conditions comfortably meets the criterion of K_{eff} <0.98.

31 The system will remain sub-critical under foreseeable accident condition by virtue of the geometric control asserted by the rack design.

4.1.2 Findings

32 I am satisfied that the safety case which EDF and AREVA have submitted for the new fuel racks is adequate.

4.2 Fuel Pond

4.2.1 Assessment

33 ND expects that criticality safety will be controlled through engineering features built into the design. For fuel storage in its simplest form geometrical constraints could maintain sufficient separation between fuel elements to prevent significant nucleonic interaction: The pressures of economy lead to requirements to optimise the number of fuel elements accommodated in a given space in the UK EPR without compromising safety.

34 EDF and AREVA have chosen to design the cells to be manufactured from borated stainless steel: the boron will absorb neutrons thereby reducing neutron interactions. Borated stainless steel is recognised as a suitable material for this purpose but quality assurance procedures will be important to ensure that the racks are fabricated from the materials of the correct specification. The racks will be expected to perform their duty of restraint and neutron absorption for a period exceeding the sixty years operational life of the reactor.

35 It will be important that the borated stainless steel should meet the specification implicit in EDF and AREVA's calculations and that the performance of the material should not be compromised by degradation over the period of its use. Although borated stainless steel is stable under the foreseen conditions it is important that that there should be periodic

monitoring to give confidence that there is no significant degradation in the neutron absorption capabilities of the racks.

36 The calculated neutron multiplication factor (K_{eff}) for all positions containing unirradiated fuel of maximum enrichment (5w% ^{235}U) and the pond filled with pure water satisfies the criteria of K_{eff}<0.98 (Ref. 136). In addition GRS has calculated (Ref. 53) that the system will satisfy K_{eff} <0.95 provided that the pond water contains a minimum of only 200ppm of boron.

37 The EDF and AREVA UK EPR will use enriched boron but the criticality calculations have been performed modelling the boron content of the pond water as natural boron, which has a proportion of ^{10}B about twenty percent - the remainder being ^{11}B which has no significant neutron absorption properties in the thermal energy spectrum; this approximation is reasonable provided that the required boron content is expressed in terms of ^{10}B.

38 EDF and AREVA have identified a number of fault scenarios which are consistent with experiences of pond storage. EDF and AREVA's analysis of a fuel element 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 stainless steel racks.

39 In each fault scenario K_{eff} for the system is shown to be <0.98 by some margin taking credit for the presence of 2000ppm of boron. The results of the calculations are summarised in Table A1 below:

Table A1: K_{eff} for Fault Scenarios

Fault	K_{eff} (including uncertainties)
Total loss of boron	0.97222
Dropped Element	0.78998
Misplacement of Element outside the racks	0.78036

40 The calculations have been performed with the Monte Carlo code MCNP4 and GRS's detailed review has confirmed that the systems have been modelled with due regard to conservative representation of manufacturing tolerances such as fuel pellet density and borated steel thickness and with full density water (shown by scoping calculations to be the worst case). Further unquantified safety margins exist by virtue of:

- the conservative omission of the burnable poison from the modelling of the fuel elements;
- modelling of the most reactive configuration of fuel assembly (3 pins missing); and
- the reduction in reactivity which results from irradiation of PWR fuel.

41 While dissolved boron plays an important part in maintaining sub-criticality, the design relies upon its presence only in accident conditions and even then a relatively low concentration is sufficient to ensure sub-criticality. This accords with IAEA guidance (Ref. 132) which cautions against taking the presence of soluble boron into account for normal operations.

4.2.2 Findings

42 I found EDF and AREVA's analysis to be comprehensive, rigorous and well structured. Reassuringly the design appears to have been developed on the basis of the analysis rather than the analysis having been developed to justify the safety of a chosen design.

43 As noted above it is important to be assured of the specification of the borated stainless steel at construction and through the life of the plant. These requirements will be followed up through Assessment Findings **AF-UKEPR-RP-A18** and **AF-UKEPR-RP-19** respectively:

> ***AF-UKEPR-RP-18:*** *The licensee shall take steps at the construction stage to assure the presence of borated stainless steel in the fuel pond storage racks in accordance with the design intent.* Milestone: This shall be complete before fuel is allowed on-site.
>
> ***AF-UKEPR-RP-19:*** *The licensee shall establish systems to monitor the borated stainless steel in the fuel pond storage racks over the lifetime of the plant so as to identify and quantify any degradation.* Milestone: This shall be complete before fuel is allowed on-site.

44 The continuing presence of boron of the correct enrichment is necessary to protect against accident conditions. A similar need has been identified through the assessment of reactor chemistry (Ref. 79) and the fuel and core design (Ref. 133) This will be pursued through Assessment Finding **AF-UKEPR-RP-20:**

> ***AF-UKEPR-RP-20****: The licensee shall establish systems to control and verify the enrichment of the boron used in the fuel pond and its continued presence in the fuel pond during its operation.* Milestone: This shall be complete before fuel is allowed on-site.

4.3 Overseas Regulatory Interface

45 There have been no exchanges with overseas regulators on this topic.

4.4 Interface with Other Regulators

46 Ultimately fuel will be despatched from the site to a disposal facility. Such transfers will be regulated by the Dept for Transport, with whom ND have liaised during GDA, but there are no particular aspects of this assessment which will present novel challenges for transport.

47 The Environment Agency will regulate the disposal of spent fuel and so has an interest in its storage. There is nothing arising from this assessment which might necessitate a change in storage conditions or pond capacity.

4.5 Other Health and Safety Legislation

48 The Ionising Radiations Regulations 1999 and the Construction (Design and Management) Regulations 2007 (Ref. 130) are both relevant to criticality safety of the

design of the EDF and AREVA UK EPR and consideration of these has been included in my assessment where appropriate.

5 CONCLUSIONS

49 This is the final report presents the findings of the criticality assessment on HSE's GDA work for the EDF and AREVA UK EPR.

50 The fuel handling and storage systems have been designed with due regard to criticality safety. Systems have been modelled with conservative assumptions with regard to manufacturing tolerances and degradation.

51 To conclude, I am satisfied with the claims, arguments and evidence laid down within the Consolidated GDA Safety Submissions for criticality safety. I consider that from a criticality safety view point, the EDF and AREVA UK EPR design is suitable for construction in the UK. There are no GDA Issues arising from this assessment.

5.1 Key Findings from the Step 4 Assessment

52 The double contingency principle has been appropriately employed.

53 The presence of boron in solution and in steel are important factors in maintaining sub-criticality and its specification and continuing presence should be verified during at construction and during operation.

5.1.1 Assessment Findings

54 I conclude that the following Assessment Findings (listed in Annex 1 of the main report) should be programmed during the forward programme of this EDF and AREVA UK EPR as normal regulatory business:

> ***AF-UKEPR-RP-18:*** *The licensee shall take steps at the construction stage to assure the presence of borated stainless steel in the fuel pond storage racks in accordance with the design intent.* Milestone: This shall be complete before fuel on-site.
>
> ***AF-UKEPR-RP-19:*** *The licensee shall establish systems to monitor the borated stainless steel in the fuel pond storage racks over the lifetime of the plant so as to identify and quantify any degradation.* Milestone: This shall be complete before fuel on-site.
>
> ***AF-UKEPR-RP-20:*** *The licensee shall establish systems to control and verify the enrichment of the boron used in the fuel pond and its continued presence in the fuel pond during its operation.* Milestone: This shall be complete before fuel on-site.

5.1.2 GDA Issues

55 I conclude that there are no GDA Issues in respect of Criticality Safety.

Table A2

Relevant Safety Assessment Principles for Criticality Safety Considered During Step 4

SAP No.	SAP Title	Description
EKP.1	Inherent safety	The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
EKP.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