Office for Nuclear Regulation

An agency of HSE

Generic Design Assessment – New Civil Reactor Build

Step 4 Fuel and Core Design 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^{TM} 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^{TM} reactor.

EXECUTIVE SUMMARY

This report presents the findings of the Fuel and Core Design assessment of the UK EPR reactor undertaken as part of Step 4 of the Health and Safety Executive's Generic Design Assessment. The assessment has been carried out on the Pre-construction Safety Report 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 Generic Design Assessment Step 2 the claims made by EDF and AREVA were examined, in GDA Step 3 the arguments that underpin those claims were examined.

The scope of the Generic Design Assessment 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 Steps 2 and 3, and to make a judgement on the adequacy of the Fuel and Core Design information contained within the Pre-construction Safety Report and supporting documentation.

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

My assessment has focused on:

- aspects of the fuel and core design which could conceivably cause the Critical Heat Flux to be exceeded and therefore impair cooling of the fuel;

- design criteria which during Step 3 appeared not to meet UK safety objectives or modern standards;

- areas of the design that introduce novel features; and

- parts of the topic area not considered in detail in Generic Design Assessment Step 3 including the validation of key computer models.

The summary of my assessment is given in this report. From my assessment I have determined that:

- Analysis presented during Step 4 has demonstrated that fuel distortion and fouling of the cladding with crud, will be limited and will not significantly erode safety margins.

- Previously identified shortfalls in safety criteria to prevent fuel clad cracking in power transients have been rectified.

- Analysis has shown that, with changes to the reactor control system, the plant is able to mitigate the consequences of more frequent faults and to avoid over stressing the fuel cladding, while at the same time retaining the reactor protection system in reserve.

- The fuel proposed for loading is a variant of the fuel loaded at Sizewell B and the development of the design has been one of evolution based on experience in existing reactors.

- The fuel design has features which increase the margin to safety limits as the fuel reaches its limiting irradiation and the cladding material performs well, with low corrosion and hydrogen uptake. Relevant experience does not indicate that there will be any significant safety issues.

- Computer models used for the analysis are generally well documented and substantiated by experimental evidence. My contractors carried out independent calculations for the sample core designs and achieved reasonable agreement with EDF and AREVA predictions.

An acceptable case has been made for loading AREVA fuel into the UK EPR reactor. However, Nuclear Directorate will need to assess the additional information that becomes available as the Generic Design Assessment Design Reference is supplemented with additional details on a site by site basis.

There are some areas where Nuclear Directorate will need additional information to underpin my conclusion and these are identified as Assessment Findings to be carried forward as normal regulatory business. These are listed in Annex 1.

I have examined the defined limits within which the fuel will operate and have required some additional constraints. I am now satisfied that these are appropriate, although attention will need to be given to ensuring that the assumptions on which the safety case is based are realised in practice. In this respect, there is a need to review actual core loading patterns and to observe the condition of irradiated fuel during core reloads. The implications of fuel assembly distortion and the propensity of the coolant to deposit crud on the fuel are two particular areas that need to be monitored.

In a number of areas, research is underway to further justify proposed operational limits and the results of these work programmes need to be assessed at the appropriate times. This includes analysis of the results of tests under fault conditions in the CABRI reactor and studies of the properties of irradiated cladding in conditions typical of dry storage.

I have sampled the proposed uranium oxide fuel loading pattern intended for a reload frequency of eighteen months. Other designs are also detailed in the Pre-construction Safety Report and these have been examined briefly. However, the design selected will need to be justified as low as reasonably possible. Should EDF and AREVA choose to load mixed oxide fuel, this would need further detailed consideration.

Overall, based on the sample undertaken in accordance with Nuclear Directorate procedures, I am broadly satisfied that the claims, arguments and evidence laid down within the Pre-construction Safety Report and supporting documentation submitted as part of the Generic Design Assessment process present an adequate safety case for the generic UK EPR reactor design. I consider that from a Fuel and Core Design view point, the EDF and AREVA UK EPR design is suitable for construction in the UK subject to assessment of additional information that becomes available as the Generic Design Assessment Design Reference is supplemented with additional details on a site-by-site basis.

LIST OF ABBREVIATIONS

AFCEN	Association Française pour les règles de conception, de construction et de surveillance en exploitation des matériels des chaudières électro-nucléaires
ALARP	As Low As Reasonably Practical
ASME	American Society of Mechanical Engineers
CHF	Critical Heat Flux (for departure from nucleate boiling)
CSNI	Committee on the Safety of Nuclear Installations
DHC	Delayed Hydride Cracking
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EBS	Extra Boration System
EA	The Environment Agency
EPRI	Electrical Power Research Institute
GDA	Generic Design Assessment
HSE	The Health and Safety Executive
IAEA	International Atomic Energy Agency
KONVOI	A particular Siemens reactor design
LOCA	Loss-of-coolant Accident
MDEP	Multinational Design Evaluation Panel
ND	The (HSE) Nuclear Directorate
NEA	Nuclear Energy Agency
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation
PCI	Pellet-clad Interaction (including both stress and corrosive effects).
PCMI	Pellet-clad mechanical Interaction
PCSR	Pre-construction Safety Report
PSI	Paul Scherrer Institute
PWR	Pressurised Water Reactor
RAPFE	Radial-averaged Peak Fuel Enthalpy
RCCA	Reactivity Control Cluster Assembly (control rod assembly)
RCSL	Reactor Control Surveillance and Limitation
RIA	Reactivity Insertion Accidents
RO	Regulatory Observation
SAP	Safety Assessment Principle
SCC	Stress-corrosion Cracking
SSC	System, Structure and Component

LIST OF ABBREVIATIONS

STUK	Finnish Radiation and Nuclear Safety Authority
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query
US NRC	United States Nuclear Regulatory Commission
WANO	World Association of Nuclear Power Operators

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1 INTRODUCTION

1 This report presents the findings of the Generic Design Assessment (GDA) Step 4 Fuel and Core Design assessment of the UK EPR reactor Pre-construction Safety Report (PCSR) (Ref. 13) and supporting documentation provided by EDF and AREVA under the Health and Safety Executive's (HSE) GDA process. The approach taken was to assess the principal submission, i.e. the PCSR, and then undertake assessment of the relevant documentation sourced from the Master Submission List (Ref. 14) on a sampling basis in accordance with the requirements of ND Business Management System (BMS) procedure AST/001 (Ref. 2). 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), Regulatory Observations (RO) were issued and the responses made by EDF and AREVA assessed.

3 Details of the assessment strategy are given in Section 2. 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. See Section 2.3 for the particular case of Fuel and Core Design.

4 A short overview of the safety case presented in the Fuel and Core Design topic area is given in Section 3 and my assessment of the case is detailed in Section 4.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR FUEL AND CORE DESIGN

5 The intended assessment strategy for GDA Step 4 for the Fuel and Core Design 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 (Ref. 1). This is summarised below.

2.1 Assessment Plan

6 The plan for assessment set out in Ref. 1 placed particular focus on the evidence required to support the values for safety limits presented as design criteria in the safety case. The assessment focused on the following topics:

- Because of its high safety significance, aspects of the fuel and core design which may influence the critical heat flux and therefore impair cooling of the fuel.

- Design Criteria which during Step 3 appeared not to meet UK safety objectives or modern standards.

- Parts of the topic area not considered in detail in GDA Step 3 including the validation of key computer models.

7 The specific Fuel and Core Design assessment aims for GDA Step 4 are detailed in Table 1. The major items in Table 1 form the basis of the assessment detailed in Section 4 of this report together with the findings from the assessment carried out in GDA Step 3.

2.2 Standards and Criteria

8   The standards and criteria that are used to judge the UK EPR are the 2006 HSE SAPs for Nuclear Facilities (Ref. 4). In particular, the following are considered:

- Key principles EKP.1 to EKP.3.
- Safety classification and standards ECS.4 to ECS.5.
- Reliability claims ERL.1 to ERL.3.
- Commissioning ECM.1.
- Maintenance, inspection and testing EMT.1 to EMT.2.
- Ageing and degradation EAD.1 to EAD.2.
- Integrity of metal components and structures EMC.1 to EMC.3.
- Reactor core ERC.1 to ERC.4.
- Criticality safety ECR.1 to ECR.2.
- Fault analysis FA.4, 9, 17, 18, 19, 20 and 21.

9   More details of these criteria are found in Table 2.

10   EDF and AREVA have assessed the safety case against their own design requirements. The French Association for Exploitation of Nuclear Fuel (AFCEN) publish design and construction rules, which constitute a set of design and inspection criteria for the fuel (Ref. 18). EDF and AREVA have adopted these.

11   Stress analysis of structural components has mostly been carried out against limiting stress requirements which appear to accord with the relevant American Society of Mechanical Engineers (ASME) code. This follows standard practice in the industry.

12   Detailed design rules are discussed in Section 3 below.

2.3 Assessment Scope

13   For the purposes of GDA, the assessment has concentrated on examining the core designed for an 18-month reload cycle utilising enriched uranium-dioxide pellets. This is because, I understand, this design is most likely to be loaded initially. In practice, actual core designs vary in detail and require some assessment prior to each core loading. The design selected will need to be justified as reducing risk to levels low as reasonably practical (ALARP) in a suitable safety case.

14   Information on mixed plutonium and uranium oxide fuel is presented in the PCSR. This has been excluded from consideration within GDA and would require a considerable assessment effort to address should this be required.

2.3.1 Findings from GDA Step 3

15   The GDA Step 3 report identified a number of specific issues which need addressing by EDF and AREVA in sufficient time to be assessed in Step 4:

- The limits and conditions proposed for the Fuel and Core safety technical specifications.
- Proposals to demonstrate no clad failures due to thermal stress in postulated frequent faults.

- Justification of design criteria and interface parameters including justification of the radial-averaged peak fuel enthalpy (RAPFE) criterion to reflect good practice.

- The case for operation with surface crud on the fuel.

- Implications of crud for the critical heat flux (CHF) and the proposed measures for surveillance.

- CHF performance of the fuel adjacent to the edge of the assembly.

- Long-term storage (up to 100 years) of the fuel following discharge from the reactor building into the onsite storage facility.

16 In each of these areas, EDF and AREVA have made substantial progress within GDA Step 4 and the detailed findings of my assessment are discussed in Section 4 of this report.

2.3.2 Use of Technical Support Contractors

17 Technical support contractors have been used in five areas:

- The development of an independent nuclear physics model of the UK EPR reactor core and the determination of reactor core kinetics parameters.

- The assessment of the flow field at the edge of the fuel assembly in the case where fuel spacer grids of adjacent assemblies made contact.

- The assessment of the fuel behaviour in the large loss-of-coolant accident.

- The assessment of crud mitigation.

- The review of the requirements for long-term storage of spent fuel.

18 The contractor review of spent fuel was managed in the Waste and Decommissioning area and is reported in (Ref. 34). I used the report as a starting point for my assessment of the likely degradation of the fuel during storage. See Section 4.13. The assessment of waste storage generally is reported in Ref. 66.

19 Similarly, the analysis of primary chemistry to assess the likelihood of crud deposits has been managed by my chemistry colleagues and is reported in Ref. 67. This reference provides some independent confirmation of the claims made. The chemistry assessment for GDA is reported in Ref. 69.

20 The remainder of these tasks were confirmatory calculations carried out using independent analysis codes.

21 The reactor core model developed by my contractor is reported in Ref. 65. The model was principally developed for use in fault studies and was intentionally not as spatially detailed as the EDF and AREVA model, but results were consistent with those of EDF and AREVA. Sensitivity studies indicated that the novel heavy reflector needed to be carefully represented. EDF and AREVA have provided additional information on their representation of this feature. This has satisfied me that they have taken reasonable measures to address the issue.

22 The issue of CHF at the assembly edge became significant when EDF and AREVA produced calculations that indicated a potential concern and this continued to be significant until experimental data was found by EDF and AREVA to quantify the effect on safety margins. In the event, my contractor had technical difficulties in completing this

work and detailed reporting has not been performed, but interim results are qualitatively consistent with those from EDF and AREVA.

23   The analysis of the effect of the large loss-of-coolant accident was prompted by the claim that very few fuel pins would be expected to burst following depressurisation of the reactor. This conclusion is welcome, but merited confirmation. Preliminary analysis work confirmed the result - giving very similar predictions to EDF and AREVA. I therefore did not require the fault to be examined in detail by my contractor.

2.3.3   Cross-cutting Topics and Integration with other Assessment Topics

24   The only area formally identified as a cross-cutting issue is the potential for the introduction of a volume of unborated water into the core in a fault transient. This is discussed briefly in Section 4.16. It is addressed in Ref. 60. In addition to this, my assessment has been integrated with that of other relevant specialists:

25   The storage of spent fuel has required collaboration. My colleague in waste disposal assessed the fuel storage and disposal facilities and the strategy. I assessed the fuel rod performance limits that need to be respected. For the details of the assessment of waste storage, please see Ref. 66.

26   Fuel crud is principally a chemistry issue and I have collaborated with my chemistry colleagues in this area. They have carried out a thorough assessment of this technically challenging area. My concern has been to ensure that there are inspection standards and mitigation measures necessary to ensure that crud will not adversely affect the fuel performance. Assessment of wider issues related to crud is found in Ref. 69.

27   The interaction with fault studies has inevitably been routine and the two assessment areas have been very closely integrated, with contact on a daily basis. My particular concern has been to ensure that the assumptions on fuel performance made in fault studies are realised in practice, both in the physical design of the fuel and in the design of core loading patterns.

2.3.4   Out of Scope Items

28   Although information has been presented for a number of fuel loading patterns, the only pattern that has been considered in detail as part of this assessment is the 18 month design with conventional uranium dioxide fuel. In particular, the use of mixed oxides of uranium and plutonium has not been assessed. It is anticipated that a particular safety case will be developed for each proposed core loading pattern and that this will be considered as a modification to the generic safety case in accordance with the arrangements defined in the site license. Operational documentation and certain aspects of the design are yet to be completed and these will be assessed during the licensing phase. More detail is found in Ref. 76.

3   EDF AND AREVA'S SAFETY CASE

29   The safety case for the fuel is set out in Chapter 4 of the PCSR and its supporting references.

30   The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids, guide tubes and top and bottom end fittings. The fuel rods consist of uranium or MOX (uranium plus plutonium) oxide pellets stacked in an M5™ cladding tube, plugged and seal welded to encapsulate the fuel. After the fuel matrix itself, the cladding and end plugs provide the second barrier to the release of radio-nuclides into the environment.

31   The square bundles of fuel rods are known as fuel assemblies. Each fuel assembly is formed by a 17x17 array, made up of 265 fuel rods and 24 guide tubes. The fuel assemblies are arranged within the reactor pressure vessel in a volume that is approximately a cylinder. This volume is termed the reactor core.

32   The fuel assembly is held together by a series of spacer grids welded at intervals to 24 tubes running the length of the assemblies. These tubes are termed "guide tubes" because, apart from forming the backbone of the fuel assembly, they act as guides for Rod Cluster Control Assemblies (RCCA) and instrumentation. They can also accommodate neutron source rods. Guide tubes that do not contain one of these components are fitted with plugs to limit the fraction of the coolant flow which bypasses the fuel.

33   The guide tubes are joined at their ends to stainless-steel manifolds termed the top and bottom nozzles. The fixings are made in such a way as to permit handling of the fuel assembly and the removal and replacement of fuel rods as required.

34   The overall structure is designed to be stiff enough to withstand substantial hydraulic forces during operation without unacceptable levels of vibration or distortion.

35   Periodically the fissile content of the fuel becomes depleted to such a level that some fuel assemblies need to be discharged for disposal and the core reloaded with a batch of fresh fuel.

36   For core reloads, the number and the characteristics of the fresh assemblies depend on the desired reactor operating parameters and the fuel management strategy. A number of proposed core loading distributions have been examined by EDF and AREVA. The operating conditions have been analysed to ensure that the fuel assembly endurance limits are respected.

3.1   Reactivity Control

37   The coolant contains soluble boron as a neutron absorber (poison). The boron concentration in the coolant is varied as required to make relatively slow reactivity changes, including compensation for the effects of fuel burnup.

38   Faster changes and control of the core axial power distribution are made by movable neutron absorber rods as part of RCCAs.

39   Additional neutron poison (gadolinium), in the form of burnable-poisoned fuel rods, is used to establish the required initial core reactivity and power distribution.

40   The reactor is shut down (tripped) by interrupting the electrical supplies to the RCCA drive motors. This causes the RCCAs to drop by gravity into the fuel assemblies and suppresses the neutron chain reaction.

41   The nuclear design basis is that, with at least a 95% confidence level:

- Fuel linear power density at the limiting location is not greater than the design limit under normal operating conditions.

- Under abnormal conditions, including the maximum overpower condition, the fuel peak power will not cause the fuel to reach the melting temperature.

- The fuel will not operate with a power distribution that violates the departure from nucleate boiling (DNB) design basis under category 1 and 2 events, including the maximum overpower condition.

- Fuel management will be such as to produce rod powers and burnups consistent with the assumptions used in the fuel rod mechanical integrity analysis.

3.2   Design Requirements and Criteria

42   The safety functions provided by the fuel assemblies are:

- Control of core reactivity and safe core shutdown whatever the circumstances.

- Residual heat removal through preservation of a coolable geometry.

- Containment of radioactive materials, in particular fission products.

43   Detailed functional requirements have been derived for each component, and thermo-mechanical design reports define specific Design Criteria. For example, the design rules for the fuel pin were set out in Ref. 56.

44   Fabrication and examination operations will be in accordance with design and construction rules for fuel assemblies of nuclear plants set out in Ref. 18.

3.3   Objectives of the Nuclear and Thermal-Hydraulic Design Analyses

45   The nuclear design analyses establish physical locations for the control rods and burnable poison rods, and physical parameters such as fuel enrichments and boron concentration in the coolant.

46   The nuclear design evaluation established that the reactor core has inherent characteristics which, together with the reactor control and protection systems, provide adequate reactivity control even if the highest reactivity worth RCCA is stuck in the fully withdrawn position.

47   The design also provides for inherent stability against radial and axial power oscillations, and for control of axial power oscillation induced by control rod movements.

48   The thermal-hydraulic design analysis and evaluations establish coolant flow parameters which ensure adequate heat transfer between the fuel cladding and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution, and mixing.

49   The mixing vanes incorporated in the fuel assembly spacer-grid design induce additional flow mixing between the various flow-channels within a fuel assembly, as well as between adjacent assemblies.

50   Instrumentation is provided within and outside the core to monitor the nuclear, thermal-hydraulic, and mechanical performance of the reactor, and to provide inputs to automatic control functions.

3.4   **Structure of the Supporting Documentation**

51   Chapter 4 of the PCSR is divided into five sections:

- Section 4.1 provides a general overview of the case.

- Section 4.2 details the fuel system design, giving the functional requirements and design requirements. It makes reference to the design evaluations which report the results of the design substantiation analysis.

- Section 4.3 details the nuclear design, giving a summary of the proposed core design and performance. It makes reference to the details of the core design and to the substantiation of the methods.

- Section 4.4 details the thermal design, giving a summary of the thermal design criteria and the fuel thermal performance. It makes reference to the detailed substantiation of the design.

- Section 4.5 gives an overview of the means of reactivity control and the provision of adequate methods of shutdown.

52   Detailed claims and arguments made are discussed below for each topic sampled along with the associated assessment.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR FUEL AND CORE DESIGN

53   My assessment has been carried out on a targeted basis and has sampled a number of topics which I believe to be important to ensure safe design and operation of the reactor core.

54   I have concentrated my consideration on areas where the UK EPR design has introduced changes or where experience has shown that particular attention is required.

55   For each topic, a brief statement of my understanding of the proposed safety case is presented below, followed by my assessment.

56   The topics address the establishment of the core performance parameters, safety limits for the fuel, and novel features of the design.

4.1 Core Power Distribution

57   Safety analysis limits generally address the most limiting fuel in the reactor and predominantly this will be fuel operating at or near to the highest local power level. Safety therefore requires that the peak linear rating be limited. There are various ways of doing this, while still meeting the economic requirements of the fuel cycle. The approach currently proposed has been assessed, although I recognise that this could be changed by any potential licensee. The change would be managed in accordance with the site license arrangements and ideally its impact on the safety case would be limited. I have therefore attempted to ensure that the generic features of the design are documented in such a way that consistency between core design and fault studies can be maintained.

58   I recognise that developing a core design is not always a simple task. For example while it may be desirable to increase margins to safety limits in potential faults, this desire may conflict with the requirements of SAP RW.2, which seeks to minimise the quantity of nuclear waste produced. Here, I believe that the principle of ensuring that the overall risk is ALARP would apply.

59   In selecting a core loading pattern, I believe that the key safety principle EKP.1 applies: "The underpinning safety aim for any nuclear facility should be an inherently safe design". For example; as far as reasonably practical, the core design should not allow the possibility that the power can increase as the core heats up. My assessment has been measured against this objective.

4.1.1 EDF and AREVA's Case

60   A number of different core design strategies are presented in Ref. 20. These comprise two conventional enriched-uranium loadings of fuel (irradiated for 18 and 22 months respectively between refuelling outages) and a mixed oxide fuel design - aimed at 18 months.

61   The medium term strategy is to load fresh fuel predominantly in a ring near the edge of the core and to shuffle it in toward the centre in subsequent cycles of irradiation; finally returning some assemblies to the very edge to complete their irradiation.

62   The required interval between outages is achieved by loading fresh fuel assemblies of the highest practical level of enrichment and limiting the initial core reactivity by the addition of material to absorb neutrons. This design uses a combination of enriched boric acid in the coolant and a limited number of fuel rods doped with gadolinium oxide.

63   The boric acid starts at a level designed to achieve a suitable core kinetic performance and is steadily reduced as the fissile material is depleted from the fuel. The effect of the gadolinium reduces naturally as it absorbs neutrons and becomes depleted.

64   The provision of a heavy steel reflector at the edge of the core helps to minimise the leakage of neutrons from the core and to maintain a reasonably uniform power distribution.

65   The power peaking is constrained to ensure that the level of boiling is sufficiently low to ensure that local vapour generation does not result in coolant void fractions associated with operational problems and that the limiting fuel assemblies are compliant with the assumptions of the fault studies.

66   The core is designed to ensure that the control rods have sufficient ability to absorb neutrons to ensure that the chain reaction can be effectively shut down in the event of a fault.

4.1.2   Assessment

67   The core design is notable because it takes fresh fuel of 5% U235 - the highest fissile isotope enrichment currently routinely loaded in commercial Pressurised Water Reactor (PWR). The limit of 5% U235 is routinely imposed to ensure that there can not be a risk of a criticality accident in an isolated fuel assembly because the quantity of U235 is insufficient.

68   In Japan there is some consideration of exceeding the 5% enrichment level by taking credit for fuel poisoning, but this has not been the practice in commercial plant in Europe. However, the move to higher enrichment is consistent with trends in the industry and provided it is done safely, will minimise some of the waste generated. Some consideration of criticality is given below in Section 4.4, but criticality during fuel storage is considered in the Radiological Protection subject area (Ref. 66).

69   The core loading patterns proposed the placement of fresh fuel close to the edge of the core where neutron leakage from the core helps to limit its contribution to the core reactivity (the alternative is to load a checkerboard of fresh and spent fuel).

70   The proposed loading pattern minimises the use of gadolinium (which would leave a residual poisoning) and is therefore beneficial for fuel utilisation and reducing the amount of nuclear waste produced. It also leads to a relatively flat power distribution, which helps to maintain margins to safety limits.   Apart from the initial core loadings, the gadolinium has been used sparingly and has a relatively modest effect on the rate of change of core reactivity.

71   This loading strategy led me to examine the effect of loading fresh fuel contiguously. In a number of cases, the loading of contiguous regions of fresh fuel has resulted in heavy crud deposits on the peripheral rods, causing fuel failures in operation. The use of the proposed loading pattern, in combination with high levels of fuel enrichment, merits particular attention.

4.1.2.1 Assembly Edge Effects

72   The assembly edge differs from the bulk of the fuel in two ways:

- the edge spacer grid design differs hydraulically from that of the remainder of the assembly; and

- the geometry of the inter-assembly gap can affect the reaction rates in peripheral pins.

73   Historically, the neutronic design of reactor cores has only focused on the bulk of the assembly. This is partly because core designs are validated by comparing predictions against neutron flux measurements taken in the instrument tube at the centre of the assembly. However, in recent years, fuel suppliers and utilities have become more aware that while resident in the core, fuel distorts subtly and the gap between fuel assemblies can vary away from the design value (Ref. 24). This causes a very local variation in the fuel-to-moderator ratio and hence the spectrum of neutron energies. Basically, as fuel assemblies move apart, the concentration of thermal neutrons in the gap increases and so does the power in peripheral pins.

74   The effect of power variation is partly compensated by the associated local increase in coolant flow rates caused by the larger gap, but as gaps become larger, the net effect is that the margin to safety limits is locally eroded. In RO 50, I asked for a detailed analysis of this effect for the UK EPR core.

75   The effect is only relevant under conditions where peripheral pins are limiting in terms of fuel rating, so the assembly power distribution needs to be considered. The power distribution is dependent on the distribution of control-rod guide tubes and poison pins within the assembly.

76   The designs of core presented in the PCSR use relatively few gadolinium pins and therefore there is a tendency for the peak rated pins to be well within the assembly. Even so, there are some notable exceptions where the peak rating is at the edge of the assembly and an allowance has been made for this effect in assessing the safety margins. This means that it remains important to place limits on fuel distortion in order to preserve margins to safety limits.

77   EDF and AREVA have taken measures to understand the distortion of their fuel during irradiation and have achieved some notable success in predicting the distribution of distortion in their cores (Ref. 25). This suggests not only an understanding of the mechanisms causing distortion, but also a consistency in fuel manufacture. Their analysis demonstrates that the levels of distortion expected for UK EPR is slightly less than for a French N4 nuclear plant and are expected to be modest (Ref. 25). The impact of the effect on the potential safety margins in faults is examined in Ref. 59. A review of the fault studies demonstrates that margins to fuel failure safety limits are either retained or the increment in the number of expected pin failures is modest and within the design criterion.

78   EDF and AREVA have taken a number of measures to limit the distortion of their fuel and further design changes are expected before fuel load. They also accept the need for monitoring fuel assembly distortion against criteria designed to ensure compliance with safety analysis. This requirement will be informed by the results of comprehensive measurements of the first core offloaded from Flamanville 3.

4.1.3   Findings

79   On the basis of the above, I am satisfied that EDF and AREVA have taken all reasonably practical measures to ensure a satisfactory core power distribution.

> **AF-UKEPR-FD-01 -** *The licensee shall, before receipt of fuel on site, review the fuel assembly measurements taken from the first core offload at Flamanville and determine the impact that the data has on the safety justification of the proposed core management.*

4.2 Core Stability

80 SAP.ECR.3 requires consideration of the stability of the core power distributions. Principally this relates to the interaction between power distribution and the distribution of xenon (which is a fission product and a strong neutron poison). Perturbations in axial power shape – usually related to moving control rods or power level - can perturb the xenon distribution and hence the local reactivity.

81 The issue of flow stability is also relevant. If power densities are high and vapour densities low, the onset of boiling can result in flow starvation. However, this is not generally a significant issue for commercial PWR.

4.2.1 EDF and AREVA's Case

82 EDF and AREVA have demonstrated, for their reference cores that xenon perturbations do not lead to loss of control of axial power shape. EDF and AREVA claim that the core designs examined are unconditionally stable to radial power oscillations and that measures are in place to limit and mitigate axial oscillations.

83 In the case of flow instability, EDF and AREVA claim that the results of CHF tests indicate that within the bounds of safe operation, no flow instability is observed.

4.2.2 Assessment

84 The UK EPR core is large compared to previous cores. However, my judgement is that the increment is marginal as argued in the next three paragraphs.

85 Axially, it is essentially part of the family of "14ft" cores and is slightly shorter than the N4 cores. There is ample experience with the N4 cores to indicate that this will not be a problem.

86 Radially, the core has an extra row of assemblies compared to N4, but the core is not much bigger than the German Konvoi plant design - which has operated successfully with a variety of core designs.

87 I view the change as incremental and I judge that the core loading pattern is more likely to be a determining factor than size alone. I think that stability is best addressed as part of a cycle-specific demonstration. Based on calculations done by EDF and AREVA for the reference cores, this does not appear to be a generic issue for the UK EPR.

88 This is an area where, to ensure plant availability, it is generally advisable to do some analysis for each proposed core load to ensure that the control rod worth is sufficient to allow power manoeuvres. However, the worth of the RCCA control banks in UK EPR is such that this is less likely to be the case.

89 In the case of flow instability, experimental evidence from CHF tests is convincing. Not only is the flow demonstrated to be stable in normal operation, but also stability is demonstrated for conditions likely to occur in frequent and most infrequent faults.

90 This issue has not been addressed for the case of the fully depressurised reactor core. However, experience of analysis and integral tests has shown that this is not an issue at power densities associated with decay heat levels.

4.2.3 Findings

91   I consider the evidence sufficient to demonstrate a satisfactory degree of core stability.

4.3   Fuel and Core Neutronic Performance in Normal Operation and Faults

92   EDF and AREVA generally (but not exclusively) assess the neutronic performance of the core in faults by using relatively simple parametric representations of the core macroscopic performance. Limiting values of the core kinetic data are used in fault studies to demonstrate safety margins for all potential core loading patterns that respect the boundaries of the parametric data.

93   A set of bounding data, validated by fault studies, is established in this way as the Safety Analysis Bounding Limits for use in core design. This approach allows a generic safety case to be developed, much of which does not need to be reanalysed for each core loading pattern. I required this to be defined with more clarity and raised RO 55 to achieve this.

94   A number of these key parameters embody requirements of key safety assessment principles. This includes the requirements for fault tolerance in EKP.1 and 2 and shutting down of the reactor in SAP ERC.2. I have examined key parameters to satisfy myself that they have been set to suitable values.

95   Since these parameters are confirmed for a particular core using reactor physics analysis, part of the assessment of this topic is an assessment of these physics methods.

4.3.1   EDF and AREVA's Case

96   The Neutronic Design Safety Analysis Bounding Limits are derived using a fully 3D representation of the core in the SCIENCE code package and then reported for the particular core in a suitable report – for example Ref. 19. The parametric data are used in representations of the reactor plant as a whole to predict the course of the fault transient. These data are selected to envelope the performance of expected core loading patterns and the suitability of the data is confirmed as part of the design of a particular core loading pattern.

97   In some cases, the complexity of the fault requires 3D analysis of the core. This is generally not done using nominal core performance data, but certain key parameters in the model are adjusted so as to represent the limiting state of expected core loadings. This allows the fault analysis to remain valid for future core designs provided certain key aspects of the core design can be confirmed.

98   The validity of the modelling is confirmed after each core reloading and at appropriate times during irradiation by suitable measurement and testing.

4.3.2 Assessment

99   The assessment focused on analysis code qualification and proposals for testing the core as loaded. I satisfied myself that the uncertainty allowances used were appropriate as required by SAPs FA.17-23 and ERC.4.

100   The design reports were reviewed to assess the adequacy of the arrangements for ensuring that the core designs fully conform to the assumptions made in the fault studies and that the basis for the uncertainty allowances used is adequately substantiated.

101   As part of RO 72, I requested additional documentation to ensure that the constraints on the core design (resulting from the fault study assumptions) were uniquely defined and justified. The definition of some of the parameters was apparent in Ref. 19, but I required additional information to justify the selection of the values and the associated uncertainty. I now feel that Ref. 70 satisfies this role. The implementation of these limits has been discussed with EDF and AREVA. Ref. 70 is now to be referenced from Chapter 18 of the PCSR and will form the basis of Operational Technical Specifications which will be developed during the licensing of the plant and will be used to constrain plant operation.

102   I reviewed Ref. 20 and satisfied myself that the core performance envelope is satisfactory to ensure that the dynamic response would be adequate and therefore inherently safe assured in accordance with the requirements of SAPs EKP.1 and EKP.2.

103   I have satisfied myself that there are arrangements to ensure an adequate shutdown margin in the event of a reactor trip as required by SAPs ERC.1-4. The generic limit on shutdown margin is detailed in Chapter 4 of the PCSR and Ref. 70.

104   The shutdown margin provided by the installed control rods is higher than in a typical PWR. However, this is desirable to compensate for the lack of rapid injection of borated water in the event of a transient cooling of the reactor. The design has enough rod worth to compensate for a cool down to the set point of the medium-head safety injection as required for accident mitigation. Analysis includes allowance for one stuck RCCA and calculation uncertainty.

105   There is sufficient boration available from various sources to account for xenon decay. In the long term, hold down of the core reactivity can be provided by the safety injection system or the Extra Boration System (EBS). The rate of reactivity increase as a result of xenon decay is sufficiently small to be capable of compensation by these make up systems. Generally the time required for operators to take the necessary actions is reasonable. Consideration of specific fault sequences is found in Ref. 60.

106   I am advised by chemistry inspectors that storage of suitable quantities of boric acid crystals on site to replenish these systems is not a design issue but one of routine operational chemistry. A finding has been raised in the chemistry area requiring suitable control of enriched boron (Ref. 69). I share the view that measures are required to ensure that all boric acid on site has a suitable enrichment.

107   The limiting moderator temperature response is defined in Ref. 70. This ensures that the reactor always has negative feedback when subject to changes in coolant temperature. This is necessary to meet the requirements of inherent safety and fault tolerance in SAPs EKP.1 and 2.

108   The design parameters on boron worth are set to ensure that the moderator density feedback can never cause an adverse power transient in accordance with the requirements of SAP EKP.2 which requires a benign response.

109   It is appropriate that key nuclear design parameters form the basis of plant rules as part of the arrangements under the proposed nuclear site license. These rules will be defined prior to plant operation. The process of updating the safety case to incorporate changes to the limiting conditions of operation is being addressed in Ref. 76. This involves changes to Chapter 18 of the PCSR within GDA to include the necessary reference material. The development of suitable limits is taken forward in cross-cutting GDA issue **GI-UKEPR-CC-02** (see Ref. 76). I am satisfied that this is an appropriate means to proceed.

4.3.2.1   Core Design Codes

110   The core power distribution and kinetic parameters are analysed using the SCIENCE code package. The code package is described in Ref. 28. It consists of a complex sequence of analysis codes:

- A neutron transport code, which is detailed in energy groups, but simple in geometry - supplying integral energy-group data to;

- A discrete-ordinate code, which compresses the energy spectrum into two groups and solves the neutron transport in detail within the geometry of a fuel assembly - creating homogenised reaction cross-section data for;

- A full representation of the whole core where the reactivity of each assembly type, is expressed as a spatially continuous function of irradiation and a number of key dependencies.

111   I have considered the model against the requirements of SAP FA.17. My sampling of the formulation of the modelling has been satisfactory. My assessment has drawn on my experience with the PANTHER/WIMS codes (used at Sizewell B) which I regard as an example of good practice in the UK.

112   The major dependencies necessary for the core model appear to be represented in the nuclear data. I reviewed the specification of the nuclear data in Ref. 41 and found that the Nuclear Energy Agency (NEA) makes similar approximations.

113   Pin powers are reconstructed from the full-core homogeneous solution using a combination of analytical functions and data from the discrete-ordinate solution. This is the same approach as adopted in the PANTHER code and in my experience works reasonably well.

114   Validation of the code package is principally found in Ref. 21. I have considered this against the requirements of SAP FA.18 and 21. I consider the validation work to be satisfactory. The validation has mostly (but not exclusively) been based on examination of the performance of the analysis route as a whole.   The studies have included comparison against both measurements and other codes as appropriate. These studies are done for a wide variety of conditions including: core depletions, xenon transients, rod drops at power and a number of faults. The validation package appears to cover the main features of significance and the agreement between experiment and prediction appears to be consistent with the uncertainties assumed.

115   The xenon transient response is good and the excore response reported for selected rod drop tests is excellent. Generally, comparison against core-follow data seems to show a similar fidelity to that of the PANTHER code, with similar variance between prediction and measurement.

116   The UK EPR differs from most commercial PWRs in that it includes a heavy steel structure adjacent to the reactor core so that neutrons are reflected back into the fuel.

This increases the efficiency with which the fuel is irradiated and reduces the neutron damage to the steel of the reactor vessel over the life of the plant. I examined the implications of this by questioning the justification of EDF and AREVA's ability to calculate the effect of the reflector.

117   EDF and AREVA have compared their calculations against the results of more detailed calculations made with an alternative method (Ref. 22). The results indicate that the heavy reflector may introduce a small increase in code uncertainty on the power of peripheral pins. This is considered acceptable in the context of the general uncertainties on reactor power level.

4.3.2.2   Physics Testing

118   The General principles for the Definition of Core Physics tests are set out in Ref. 30. On a sample basis, these conform to the recommendations of the relevant IAEA standard (Ref. 74) and to the American Standard (Ref. 75). The tests include verification of the core power distribution, the effectiveness of the control and shutdown system and the core dynamic response. The acceptance criteria are set at widely accepted values.

4.3.3   Findings

> **AF-UKEPR-FD-02** - *The licensee shall, before first fuel load, review the results of available EPR physics testing and confirm uncertainty allowances in the safety case.*

4.4   Core Misloading Faults

119   Considerable care is expended on designing a scheme for the placement of the fuel in the core so as to achieve the desired core power distribution (and core characteristics in general). Failing to load fuel in the proposed arrangement is a foreseeable error and requires satisfactory measures to mitigate the consequences. I therefore asked for supplementary information on the arrangements and consequences of this fault.

4.4.1   EDF and AREVA's Case

120   During the fuel loading process, a large number of successive fuel misloading errors are needed to reach criticality thanks to the loading procedure. Therefore the impact of a super critical core on the fuel has not been estimated for these conditions.

121   When each assembly is loaded, the neutron flux at the Source-Range Detector location is compared against a reference level. The alarm threshold is set to three times the reference level expected for the assembly. Therefore core misloading faults sufficiently serious to lead to criticality will be detected by the Source-Range Detectors.

122   A flux increase greater than the reference flux may be detected by the Source-Range Detector during the two or three fuel assembly insertion steps preceding the criticality. However, this depends on the fuel configuration and on the reference flux.

123   The Source-Range Detectors are therefore able to detect any misloading faults leading to a critical core and, depending on the configuration, some misloading faults before they reach criticality. The vast majority of conceivable misload faults will not lead directly to a criticality event.

124   Once the core is built, the placement of assemblies is verified visually before the reactor vessel head is replaced. Should this visual check fail low-power physics testing will detect all except relatively benign misloadings and prevent ascension to full power.

4.4.2 Assessment

125 The procedures for loading fuel have been changed following recent misloading events to ensure that omission of a single assembly from the loading sequence will not lead to a serous disruption to the loading pattern.

126 EDF and AREVA have demonstrated that a large number of misloaded assemblies are required to cause a reactivity fault prior to completion of the loading. They argue that this is too low a probability event to require detailed analysis. In essence, they claim that this fault is beyond the Design Basis. This is a strong claim and will need detailed substantiation when procedures for core loading are developed. I am therefore raising a finding to this effect.

127 In response to TQ 863, EDF and AREVA advised that the available computer codes are not designed to perform calculations of the impact of core super criticality with an incomplete core during refuelling phases. However, I note that measures are in place to mitigate the event before this occurs and these form the basis of the safety case.

128 As the loading progresses, the signal on neutron detectors is compared against the expected signal at that stage in the loading sequence. The EDF and AREVA analysis of the reactivity, after successive assemblies are added, indicates to me that the signal should permit anomalies to be identified before they lead to a significant event in a part-constructed core. However, this is not likely to detect more limited misloads that do not result in criticality.

129 More limited misloads are credible and the plant needs to be protected against operating with an adverse core power distribution.

130 Once the core is built, the placement of assemblies is verified visually before the reactor vessel head is replaced. I believe that this is potentially a strong measure for identifying misloading originating from fuel pond operations.

131 Low-power physics testing will detect all except relatively benign misloadings. These physics tests appear to be consistent with good practice in this respect. The use of a system of incore flux detectors as part of the control and protection system gives the EPR design some degree of enhanced protection against this fault at power compared to many existing PWRs.

4.4.3 Findings

132 I am satisfied that the measures taken to minimise the effect of misloading provide multiple barriers to operating with a misloaded core and are consistent with good practice elsewhere. However I feel that the protection against severe fuel misloading faults needs further justification.

> **AF-UKEPR-FD-03 -** *The licensee shall, before first fuel load, demonstrate that the procedures proposed for loading the reactor core with fuel will ensure that an uncontrolled criticality is incredible or that all reasonably practical measures have been taken to prevent this.*

4.5 Fuel Pin Performance Modelling

133 The fuel itself is fabricated as the first barrier to the release of fission products into the plant and potentially to the environment. Fuel integrity is an important part of any strategy of defence in depth as required by SAP EKP.3.

134   A set of design criteria are required to ensure that the fuel operates within its design envelope taking account of any degradation which may occur during operation. Safety margins are generally assessed by modelling the performance of the fuel pin for a postulated history of operation, including (where appropriate) fault conditions.

135   Assessment of the COPERNIC fuel modelling code is discussed below and then in the following sections particular issues that have arisen during fuel operation from other reactors are reviewed, after which limiting conditions of fuel operation are considered.

4.5.1 EDF and AREVA's Case

136   Details of the design of the fuel pin are given in Chapter 4 of the PCSR with detailed analysis results in Ref. 43.

137   The fuel pin is in most respects a standard product with literally millions of reactor years of cumulative operating experience. This ensures that the risk of unexpected problems is minimal. There have been some changes, the most notable being the addition of a lower (in addition to the upper) plenum to accommodate fission gas. This is one of the measures designed to increase inherent safety margins and hence permit better fuel utilisation.

138   The pin is modelled mostly by the COPERNIC fuel performance code. This code provides the principal analytical tool for confirming that the planned operation will not take the fuel outside its design limits (during normal operation and frequent faults).

139   The code has a series of empirical models based on a large body of experimental data. These are reported in Ref. 55.

140   Models have been developed to account for:

- the distribution of fission rate within the fuel pellets;

- the conductivity of heat to the rod surface;

- the release of fission gas from the fuel pellet into the fuel pin gas plenum; and

- the stresses and strains in the pellet and cladding.

141   These models and the associated uncertainty have been systematically qualified.

142   In the case of rapid transients, this analysis is supplemented by use of the SCANAIR code, which is discussed in Section 4.14.

4.5.2 Assessment

143   The COPERNIC model is similar to that of the ENIGMA code used for Sizewell B. It requires data from an external source to define coolant conditions and the temporal and spatial distribution of power within a fuel rod. It determines the radial power distribution within the fuel pellets, together with the temperatures and the stresses.

144   The agreement between the model of power distribution and measurement is good over the relevant range of conditions, so is the agreement between measured and predicted fuel temperatures.

145   Prediction of fission gas release has been systematically compared against the data and while there is significant scatter in the data, the uncertainties have been well qualified and both best estimate and conservative modelling has been developed.

146   I examined the gas release for fuel in the region of 50-60 MWd/kgU in particular; there was a tendency in some fuel pins to under predict the threshold temperature for fission gas release, but this was found to have an insignificant effect on safety margins because it only applied to fuel irradiated at a very low power density for which safety margins are large.

147   In response to my questions, fresh data was analysed for particularly relevant high-rated fuel and the measurements were in good agreement with predictions.

148   Prediction of clad strain resulting from irradiation-induced creep was good and the high-stress thermal creep was well predicted for the M5™ alloy, with the reservation that the COPERNIC model does not represent pellet end effects, which tend to cause higher clad stresses at the ends of the pellet.

149   In the case of Pallet-clad Interaction (PCI) analysis, the pellet end effect is equally omitted from the modelling used to derive design criteria and the analysis used to confirm compliance. The effect of this omission is therefore essentially cancelled out.

4.5.3   Findings

150   I note that the modelling in the version of COPERNIC currently under development is more three-dimensional and this is welcomed, but I consider the current model fit for purpose.

4.6   Fuel Clad Corrosion

151   In previous generations of fuel, corrosion of the cladding (and associated embrittlement) has limited the permitted irradiation of the fuel. However, the change of the material to the M5™ alloy appears to have effectively removed this restriction. Nevertheless, it is necessary to be satisfied that the issue of corrosion is effectively controlled.

4.6.1   EDF and AREVA's Case

152   EDF and AREVA rely principally on operating experience feed back from post irradiation inspection and metrology to justify the performance of the fuel cladding in reactor.

153   The data collected to date is extensive and demonstrates very good corrosion performance (Ref. 51).   However, this data has been supplemented by extensive fundamental research and by out of pile experiments in extreme conditions - above the maximum void fraction permitted in operation. Some of the operational data has been for relatively hot conditions, with levels of boiling higher than envisaged (Ref. 27).

154   Where anomalies have been observed during fuel outages, thorough investigations have demonstrated that the particular circumstances did not raise general concerns for the fuel performance, e.g. Ref. 38.

4.6.2   Assessment

155   M5™ cladding has been accepted by NII for use at Sizewell B and to my knowledge, no safety issues relating to the cladding material have arisen.

156   In order to ensure control of reactor chemistry, the void fraction at any point in the fuel assembly is constrained to be below 5%. This is justified in Ref. 27.

157   I am conscious that it is necessary to limit the level of boiling in the core to ensure that coolant chemistry is suitable. In my experience, the industry has operated with a 10% constraint on coolant void fraction and only a small number of incidents have occurred. However, evidence from Electrical Power Research Institute (EPRI) also indicates that it is necessary to limit the level of boiling to significantly below 10% to mitigate the risk of crud formation.

158   Mostly M5™ rods have been free from unexpected levels of oxidation. In the few cases where anomalies have arisen, there were operational factors which are unlikely to be repeated because the lessons learned have resulted in changes to practices. The most notable event is reported in Ref. 38 which presents a good root cause investigation and gives satisfactory assurance that repeat events are unlikely.   Having reviewed the evidence I judge that the selection of a 5% limit on coolant void fraction is conservative and helps to eliminate cladding oxidation as a safety issue for this cladding.

159   Design criteria on acceptable levels of cladding oxidation are similar to those used at Sizewell B and are considered appropriate, but are superseded by the more restrictive constraint on hydride levels required for dry fuel storage.

160   Process control detailed in the AFCEN standard is a combination of visual inspection and destructive sampling. This is consistent with my understanding of established good practice.

4.6.3   Findings

161   Overall, I am satisfied that the oxidation of M5 cladding in UK EPR is likely to be satisfactory.

4.7   Crud Mitigation

162   The corrosion of steam generator tubes results in the release of nickel and iron into the primary circuit and potentially its deposition on the fuel. This forms a crystalline layer termed crud, which becomes activated during the fuel irradiation.

163   As chemistry changes and the crud layer grows, some of the crystalline material is released from the fuel surface in the form of active particulate, which becomes distributed around the primary circuit and the fuel storage pond. This activated material can increase dose to plant operators. Furthermore, the crud itself can in extreme cases, inhibit efficient heat transfer from the fuel, leading to fuel degradation and fuel cladding failure. Crud can incorporate boron from the coolant and perturb the power distribution leading to concerns relating to the core neutronic performance.

164   For all the above reasons, it is necessary to take reasonably practical measures to limit and monitor crud formation. I issued RO 49 which asked EDF and AREVA to address this issue in Step 4 of GDA.

4.7.1   EDF and AREVA's Case

165   In Ref. 52 EDF and AREVA have proposed a surveillance scheme based on visual inspection of a targeted sample of fuel elements during refuelling outages. The fuel is categorized according to the extent and morphology of any crud into three crud categories: Light, Moderate and Heavy.

- Category 1 - Light crud - does not need corrective actions. But this weak signal needs to be looked at (e.g. water chemistry has to be checked).

- Category 2 – Moderate crud - needs strengthening of the non-failed fuel surveillance and inspection program and follow up on the fuel assemblies concerned after the next cycle. Water chemistry has to be addressed.

- Category 3 - Heavy crud - requires immediate actions in order to mitigate impact on the fuel cladding. No reloading of the concerned fuel assemblies in their present state is allowed. A detailed understanding of the crud is needed through crud scraping and thorough crud analysis campaigns. Corrective actions could be core redesign, water chemistry adjustment and ultra-sonic fuel cleaning.

166   Examination of the effect of heavy crud deposition on the corrosion of a small number of M5™ rods indicated that little effect on cladding oxidation occurs for this cladding alloy until the crud deposition becomes severe.

167   Impact of crud on thermal hydraulics has been assessed. Calculations have been performed with the 3-D subchannel code FLICA III-F. The crud deposit is modelled using an increased rod friction coefficient deduced from roughness values measured on a rod sample with a deposit representative of the Category 2 crud. The margin to safety limits was assessed using the ratio of the limiting heat flux to the design value Departure from Nuclear Bowling Ratio (DNBR).

168   The values of DNBR calculated for Category 2 crud are 1% lower and therefore EDF and AREVA do not consider that the effect is significant for safety. Fuel Assemblies which present crud classified in Category 1 or 2 will be reloaded without specific justification.

4.7.2   Assessment

169   I note that a number of measures are proposed to mitigate the risk of crud formation. These are welcome and are the subject of detailed assessment by my colleagues in the chemistry specialism (Ref. 67). My focus has therefore been to ensure that sufficient measures are in place to prevent crud leading to fuel degradation.

4.7.2.1   Effect of Crud on Fuel Oxidation

170   Ref. 52 reports the effect of crud on the oxidation of a fuel rod with a heavy crud deposit. EDF and AREVA removed the bulk of the crud and measured the underlying oxide thickness. The thickness was at the top end of what would be expected in the absence of crud and therefore had a large margin to safety limits. This observation is not conclusive given the limited size of the sample and also because of the method employed. However, the study suggests that the effect of crud on oxidation is largely restricted to severe cases of crud deposition. This indicates that, provided crud formation is limited, the fuel integrity will not be threatened.

4.7.2.2   Surveillance

171   Based on my experience of observing crud deposited on fuel at Sizewell B, the visual references included in Ref. 52 are reasonable and provided that the control measures proposed are effectively implemented, my judgement is that the controls in place will be sufficient to ensure integrity of the fuel.

4.7.3   Findings

172   Based on the arguments detailed above, I am satisfied that the measures proposed will protect the fuel against unacceptable levels of degradation as a result of crud formation.

4.8 Fuel Clad Stress

173   When fuel is loaded into the fuel pin cladding, a gap between the fuel and the cladding exists, which initially gives the cladding some protection against the effects of fuel pellet thermal transients. However, under the influence of coolant pressure and neutron flux, this gap closes during operation.

174   Following the gap closing, fuel swelling and thermal expansion of the fuel induces circumferential cladding strain. In the past, this has occasionally led to failure of the cladding in operational transients and it is considered good practice to provide protection for the cladding at least in normal operation and frequent faults. It is for this reason that I issued RO 42, which asked AREVA and EDF to justify that they have taken all reasonably practical measures to avoid cladding failure.

4.8.1 EDF and AREVA's Case

175   Design criteria are defined with the aim of minimising the risk of cladding damage:

- Analysis demonstrates that stresses never reach the yield strength of the cladding alloy and that creep strain does not exceed the material ductility.

- Protection ensures that the strain energy is never sufficient to cause failure by stress-corrosion cracking.

- Cladding fatigue is analysed based on a pessimistic analysis of the likely reactor power transients and is not expected to be significant.

176   The cladding ductility is preserved by ensuring that the cladding stress will not exceed the 0.2% strain criteria - derived from conventional tensile tests. The creep ductility is preserved by ensuring that the strain does not exceed 1%.

177   The risk of cladding failure by PCI is evaluated by comparing the rod mechanical loadings calculated with the COPERNIC thermal-mechanical code with a PCI failure criterion called the technological limit of the material. This technological limit is based upon the interpretation of a database of power ramp tests performed in test reactors. The limiting value is defined from the ramp test simulations carried out with the same thermal-mechanical code used to demonstrate compliance (Ref. 47).

178   The analysis is based on dedicated protection implemented in the reactor control system which triggers a sequence of turbine load rejection and selected RCCA insertion termed a Partial Trip.

179   The analysis method underpinning the Partial Trip is detailed in Ref 53. This approach is based on the following principles:

- Full Reactor Trip is not triggered - in order to hold post-trip systems in reserve.

- Automatic actions have been preferred to an administrative solution.

180   The analysis includes allowance for 30 days Extended Low-Power Operation prior to the fault.

181   This analysis shows that the PCI limitation system retains a safety margin to cladding failure in frequent fault transients and simplifies administrative control during return to full power after Extended Low-power Operation (Ref. 49).

182   Fuel is also in-principle susceptible to damage by fatigue loading, but in practice, analysis and experience demonstrate that this is not significant.

4.8.2 Assessment

183 The ductility of zirconium is generally high, but irradiation hardens the material and can result in a significant reduction in macroscopic ductility. It is therefore prudent to prevent material damage by limiting the amount of plastic deformation. The safety case does this by using the conventional limits on cladding stress and strain. I judge that these are appropriate. However, in addition to preserving the material ductility, analysis of stress-corrosion cracking and fatigue is required and is considered below.

4.8.2.1 Stress-corrosion Cracking

184 The protection against stress-corrosion cracking is ensured by limiting the cladding stress permitted in normal operation and frequent faults. The limit is expressed in terms of a strain-energy density evaluated using the COPERNIC fuel rod computer model (Ref. 47). I support the use of this parameter because in my experience, it provides a better discriminator for cladding failure than clad stress.

185 The criterion is termed the Pellet-cladding Interaction (PCI) Technical Limit. This is conservatively defined by calibrating the COPERNIC code against power-ramp tests on samples of irradiated fuel rods.

186 The number of ramp tests specific to M5™ cladding is limited, but the number performed on Zircaloy 4 is large and M5™ appears to perform at least as well as Zircaloy 4. Moreover, ramp tests on rods are supplemented by material tests on irradiated cladding segments. The approach to defining a criterion is similar to that used for M5™ cladding at Sizewell B and I am satisfied that the limit is justified as part of a reasonably practical approach to PCI protection.

187 The protection is based on limitation of the peak power density in the Reactor Control Surveillance and Limitation (RCSL) system (Ref. 53). The peak power density is derived from Self Powered Neutron Flux Detectors installed in certain fuel assemblies. This is potentially a significant advance over many existing reactors which rely on external instrumentation to deduce the power distribution.

188 I note that the details of the RCSL system are under consideration by specialists in control and instrumentation and these aspects will be considered by my colleagues as part of licensing. However, these considerations are not expected to significantly affect this aspect of the system.

189 When the PCI limit is reached, staggered actions are initiated:

- prevent generator power increase and RCCA withdrawal;

- reduce generator power and insert an RCCA bank; and

- Partial Trip (release of selected banks of RCCAs).

190 The purpose of the Partial Trip is to achieve a fast power reduction, while not fully shutting down the plant (which otherwise places demands on standby safety systems to meet the heat removal requirements). The safety systems remain available if required, but I support the view that such systems should be kept in reserve if safety can be achieved within the bounds of normal plant operation.

191 The transient response of the system has been modelled for an appropriate set of Frequent Faults, assuming pessimistic initial conditions. Ref. 49 demonstrates that the system can achieve effective limitation of reactor power. Allowance is made for

uncertainty in power reconstruction at a level which seems reasonable and the treatment of extended low-power operation appears to be conservative.

192 Overall, I feel that the system engineered to protect the fuel against over stressing of the cladding is an example of good practice.

4.8.2.2 Cladding Fatigue

193 The assessment of clad fatigue assumes that cladding failure occurs when the fraction of the experimentally-derived fatigue life - the cumulative damage - is equal to 1. As fatigue analyses do not take into account the effect of creep, the design criterion used stipulates conservatively that the fractional cumulative pure fatigue damage (or consumed life) must be less than 0.8 (Ref. 50). This criterion is similar to that of Sizewell B and in my experience is appropriate.

194 An assessment of the stress associated with likely power manoeuvres has been made and conformance with limits is demonstrated on a conservative basis. However, my judgement on this is that the evidence is only limited. COPERNIC 2.4 does not have a detailed 3D model of the fuel pellet and therefore only returns a pellet-centre strain. An analytical approach to this issue therefore has a degree of uncertainty.

195 The case also relies on evidence from existing EDF plant. Fuel has been irradiated while deliberately subject to intensive reactor power transients required to support changes in the demand for electricity on the French power grid. The data analysis of the recordings taken during this irradiation campaign allowed the power variations to be quantified. These tests largely bound the expected operation of the UK EPR (Ref. 26).

196 Hot cell fatigue tests on these fuel rods did not show any difference from rods which had not been cycled. This leads to the conclusion that in extreme conditions, the fatigue damage caused by grid follow is negligible. I have therefore not pursued this issue further.

4.8.3 Findings

197 I am satisfied that EDF and AREVA have proposed to take all reasonably practical measures to limit the risk of cladding failure due to thermal stress.

4.9 High Burnup Issues

198 I required that maximum fuel rods burnup be specified. The purpose of this is to ensure that new issues (which could potentially arise with increases in burnup) are adequately analysed and accommodated in the safety case. For the purpose of this safety case a limiting rod irradiation of 62 MWd/kgU has been specified (corresponding approximately to maximum Fuel Assembly burnup of 58 MWd/kgU).

4.9.1 EDF and AREVA's Case

199 EDF and AREVA argue that burnup is not a primary constraint on the fuel and is implicit in the qualification of their analysis methods. EDF and AREVA envisage an increase in fuel irradiation at a future date.

4.9.2   Assessment

200   The value of 62 MWd/kgU proposed for an irradiation limit for GDA is well within the extremities of the body of data used to validate the fuel performance code (which extends as far as 70 MWd/kgU). It is close to the boundary of the bulk of the data on fuel assembly performance. No divergence in expected, performance is apparent within this range and I am not aware of any cliff-edge effects beyond 62 MWd/kgU.

201   I know that as burnup increases, the fuel pellet becomes more porous and fission gas release rate increases, but the fuel and the core designs appear to include measures to accommodate this. The ALARP arguments for determining the optimum irradiation are complex and are not presented in detail in the submission. However, I note that the values of burnup proposed are consistent with established practice and a detailed consideration of the issues is more appropriate if increases are proposed.

202   I note that the formation of fission gas bubbles within the outer rim of the fuel stresses the fuel material and potentially affects the stability of the material in fault transients. Simulated fuel response to large LOCA has demonstrated that, under certain conditions, very high burnup fuel can experience pellet fragmentation. However, IAEA's Committee on the Safety of Nuclear Installations (CSNI) conclude: For burnups up to 60-65 MWd/kgU, it is believed that any fuel dispersal would be minimal (Ref. 15). I also note that the burnup proposed has been an established limit in the US for many years.

4.9.3   Findings

203   On this basis, I am content with the burnup limit proposed.

4.10   Critical Heat Flux (CHF)

204   Provided that the fuel cladding surface is liquid water cooled (usually by a combination of forced convection and nucleate boiling), the cladding surface temperature is never far above the boiling point of water at the local pressure. This is a necessary condition for ensuring fuel integrity.

205   As the fuel surface heat flux is increased, a critical value is eventually reached where the generation of vapour prevents sufficient water contact with the surface. This value is a fundamental design criterion for demonstrating satisfactory heat removal from the fuel in anticipated faults as required by SAP ERC.1.

4.10.1 EDF and AREVA's Case

206   In normal operation and frequent faults, integrity of the fuel requires that keeping below the CHF is achieved by maintaining sufficient water flow and limiting the local power density.

207   In infrequent faults, some degree of overheating is acceptable provided that the cladding is not damaged to the extent that a coolable geometry is lost.

208   The safety analysis examines the margin between the fuel surface heat flux and the CHF at which the nucleate boiling process breaks down. This is expressed in terms of the ratio between the limit and the local heat flux: the DNBR. The approach is semi-empirical.

209   The CHF is evaluated by conducting a series of experiments on a limited number of electrically heated pins; designed to simulate part of a fuel assembly. The results are

correlated with the local thermal conditions expressed in terms of coolant velocity and enthalpy (Ref. 33).

210   The limiting local conditions used to derive the correlation are calculated using the FLICA 3 thermal-hydraulics code. FLICA 3 is qualified for this purpose by a combination of comparison against other codes and experiment as discussed in Ref. 23.

211   The analysis takes account of various sources of uncertainty including: the uncertainty in the operating conditions; the fuel manufacturing parameters; and computer models. A statistical safety analysis limit on the DNBR is defined to encompass these uncertainties at the 95% probability level at 95% confidence.

212   The method allows a combination of the statistical and deterministic factors affecting the DNBR:

- measured thermal-hydraulic parameters;

- CHF correlation;

- design code error;

- transient versus steady state;

- rod bow penalty; and

- loss of representativeness taking into account the discrepancy between the real minimum DNBR value in the core and the DNBR value reconstructed with the simplified algorithm used for analysis.

213   Other uncertainties associated with the progression of the fault are accounted for within the models that derive the core conditions considered.

214   The statistical method is based on a Monte-Carlo approach. Uncertainty distributions are applied to the thermal-hydraulic parameters and these distributions are sampled, leading to a large number of DNBR values. Then a statistical DNBR limit, taking into account the global contribution of the uncertainties, is extracted from the results of the sample.

215   For infrequent faults, the full statistical approach is not employed. Only the CHF correlation uncertainty and the rod bow penalty are taken into account when setting the value of the design limit. The other uncertainties are included directly in the transient analysis.

216   The CHF correlation is well-behaved in its qualified range and there is no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundles.

4.10.2   Assessment

217   I have assessed the analysis against SAP ERC.1: that sufficient safety margin to fundamental safety functions should be provided. My focus has been on ensuring that the analysis appropriately represents the plant, including any variability in manufacture and changes during operation.

218   The assessment of CHF relies on the use of an appropriate code and a valid consideration of uncertainty. I gave consideration to the use of the FLICA code, but since the techniques used by this code are well established within the nuclear industry, my principle consideration has focused on the material factors that have arisen from recent operational history which may adversely influence the conditions in core. In particular, consideration has been given to the conditions at the edge of a fuel assembly for reasons given in Section 4.1.2.1. My conclusions on the use of FLICA follow.

4.10.2.1 Use of the FLICA 3 Computer Code

219 Ref. 63 gives an overview of the models and correlations employed in FLICA in accordance with the requirements of SAP FA21. It does not provide reference to the user guide for execution of the code. Suitable documentation and training will need to be provided to the licensee or his Design Authority to ensure that a satisfactory level of knowledge is maintained to enable the intelligent customer role. This is part of the motivation for the finding below.

220 The qualification of the FLICA 3 code is reported in outline in Ref. 63 which is updated for the latest version in Ref. 23.

221 The code has been used in France for CHF assessment for a long period and qualification has been reported at conferences from time to time for example Ref. 24. The development work carried out for FLICA 4 is impressive and takes it beyond the codes generally in use for CHF assessment, but EDF and AREVA have chosen to use the simpler and faster code FLICA 3, using an approximation to the transverse momentum equation rather than the full porous-medium solution in FLICA 4.

222 FLICA 3 is analogous to the COBRA 3 code - which is used for the same role for Sizewell B - but the modelling of boiling fluid has been developed in FLICA 3 and has, in my view, been enhanced slightly to give the code a potentially slightly wider range of applicability, but I judge that this will have little effect for the conditions of interest and this is confirmed by cross-code comparison.

223 I note that the modelling of transverse frictional pressure loss is relatively crude. However, in my experience, this is not significant for the conditions in which FLICA 3 has been used and therefore I have chosen not to consider this issue in detail.

224 I believe that FLICA 3 is of similar fidelity to existing codes that have been used for licensing submissions in the UK (such as COBRA). However, I note that OECD has recently carried out benchmark studies on boiling flow in rod bundles. This data will allow the fidelity of the code to be assessed with greater confidence than previously and I believe that such comparisons should be carried out in accordance with the guidance in SAP FA.24.

4.10.2.2 The CHF Correlation

225 EDF and AREVA have developed a CHF correlation that is applicable across a wide range of conditions without an apparent loss of precision.

226 At low pressure, there are data covering the range in which the correlation is likely to be used and the correlation appears to retain its accuracy across the required range.

4.10.2.3 Uncertainties

227 The approach of using a statistical combination of uncertainties in the case of frequent faults is fairly standard, although the original method developed by Westinghouse assumed a linear response surface. The Monte-Carlo approach presented is in principle more generally applicable.

228 The selection of uncertain parameters for the derivation of the Safety Analysis Limit is reported in Chapter 4.4 of the PCSR. The parameters selected and the values quoted are in line with my expectations.

229   The analysis did incorporate an allowance for fuel pin bowing as a result of irradiation creep – addressing the issue of gap closure, but did not explicitly address the issue of assembly distortion – opening gaps between assemblies. This has become a concern in recent years after a number of events where slow insertion of RCCAs has been observed. The topic has lead to the development of sophisticated analysis tools. See Ref. 24.

230   EDF and AREVA argue that no additional allowance is required for the current core loading patterns provided that the assembly distortion is not greater than expected. This is considered in more detail in Section 4.1.2.1.

231   I have considered the implications of the approximations made to represent the assembly power distribution generically, and I have requested a number of sensitivity studies. On the basis of these, I am satisfied that the approach adopted is conservative.

232   I have required additional limits on cladding temperatures be specified for infrequent faults to demonstrate that, in the event of degraded cooling, the fuel continues to maintain a coolable geometry. Details of the relevant fuel design criteria are found in Ref. 26.

4.10.3   Findings

233   Overall, I have concluded that a convincing case has been made to support the CHF assessment method used for the analysis of fuel safety margins in normal operation and faults.

> **AF-UKEPR-FD-05 -** *The licensee shall repeat the recent OECD benchmark studies on boiling flow in rod bundles by first fuel load and update the FLICA qualification documents.*

4.11   Fuel Assembly Component Design

234   The design of fuel assembly is required to provide a reliable means of locating the fuel rods and permitting their handling. The assessment therefore focuses on any areas of novelty and measures taken to improve reliability as required by SAPs ERL.1 and 2 in addition to requirements specific to the reactor core.

4.11.1   EDF and AREVA's Case

235   The fuel assembly design for UK EPR is described in Ref. 42. This is a fairly standard 17x17 rod bundle array, with a few detailed changes:

- replacement of the redundant central instrumentation tube by a fuel rod; and

- adaption of the bottom nozzle and bottom grid system to improve fretting and anti-debris performance.

236   The design requirements for the fuel assembly as a whole and the individual sub assemblies are given in Chapter 4.2 of the PCSR, with the substantiation reported in Refs 44 and 45.

237   The purpose of the assembly is to:

- maintain the dimensional stability of the fuel rod bundle and the core as a whole;

- enable fuel loading operations;

- protect the fuel against debris present in the coolant;

- ensure satisfactory levels of coolant mixing within the fuel assembly; and

- support the fuel rods without causing significant fretting damage to the cladding.

238   Satisfactory performance is assured by thermal and structural analysis against the identified design requirements, together with operational experience based on a programme of in-service inspection.

4.11.2 Assessment

239   The safety assessment principles require analysis of the safety margins throughout the life of the assembly and demonstration of tolerance of fault conditions.

240   Due to the large number of components in a reactor core, some defects are inevitable. A small number of fuel rods with failed cladding can in exceptional circumstances be tolerated, but the World Association of Nuclear Power Operators' (WANO) expressed intent is that core reloads would be expected to be defect free and to operate without failures developing. I have taken this as my yard stick of good practice.

241   I have sampled the component design documentation to determine whether the design follows established practice and to consider whether the analysis results introduce any new issues compared to established practice.

242   The design of the fuel assembly skeleton against clad stress limits is reported in Ref. 44. The criteria adopted are derived from established standards. However, I also base my judgements in this area on operational experience and accelerated integral testing.

243   The dimensional stability of the core is maintained by locating the assemblies firmly at the top and bottom and designing an intermediate structure that is adequately stiff, while minimising the material present within the active core. The design approach used for the UK EPR is not significantly different for the majority of PWR designs and consists of stainless steel top and bottom nozzles – with holes for location pegs – connected via a number of guide tubes and held down onto the lower core support plate by an arrangement of Inconel leaf springs on the top of the assembly.

244   Bowing of the assembly has already been addressed in Section 4.1.2.1. However, this is impacted by the detailed assembly design. EDF and AREVA calculate the axial growth of fuel pins and control rod guide tubes to ensure that the clearance between these components is preserved and also that the travel on the hold-down springs is not exhausted. The change to M5 for the guide tubes was intended to accommodate increased fuel irradiation, but the data presented to date has shown a degree of variability (Ref. 44). This is believed to be due to details of the particular assembly designs, but a further change in guide-tube material is anticipated before fuel loading. The evidence presented to date does not indicate a concern relating to the management of this issue.

245   Spacer grids are designed to allow a small clearance between adjacent assemblies to facilitate core loading. The data on growth of the spacer grids is limited for high burnup, but does not indicate a problem within the burnup range considered (Ref. 44).

246   The lifting and handling aspects of the fuel design are not significantly changed from previous designs and the design achieves generally satisfactory experience. Fuel lifting operations are assessed in Ref. 68.

247   The trapping of debris has been the subject of sustained development; both experimental and in product optimisation. The proposed TRAPPER® bottom nozzle is a logical development, with slightly smaller holes than the current version. I accept that it is an

improvement from the point of view of debris capture. The additional bottom grid will also provide some degree of protection.

248   The design of the fuel assembly skeleton is assessed against the loads anticipated as a result of accelerations experienced in normal operation and faults. The analysis is summarised in Ref. 44.

249   The top and bottom nozzles include complex geometric designs, intended to optimise coolant pressure loss. These designs have been subject to stress analysis using finite element methods against allowable stress criteria. The shortening of the bottom nozzle feet brings a benefit in terms of stress and stiffness over the established design.

250   The spacer grids have been subject to crush tests to determine their bucking loads. These have been compared against the predicted impact loads when the assembly is subject to Seismic and LOCA events (Ref. 44). The predicted assembly stiffness in the model has been tuned to stiffness measurements on sample fuel assemblies and the macroscopic performance of the assembly is similar to that of existing fuel. The analysis method is similar in concept to that of Sizewell B.

251   The analysis assumes a notional bounding earthquake, but would need to be reconsidered if a proposed site had specific requirements which exceeded this.

252   One omission from this analysis is consideration of the double-ended guillotine-break large LOCA. This is omitted based on the low likelihood of the fault. See Ref. 60 for further discussion of this fault. I have not pursued this issue because I believe that, in the absence of fuel clad ballooning, a case could be made that the consequences of grid spacer buckling are acceptable in the highly unlikely event of the fault occurring.

253   The hydraulic forces and pressure losses for the proposed fuel are detailed in Ref. 45. They are essentially the same as the N4 fuel and are also similar, on a component basis, to Sizewell B. The springs on the top nozzle are optimised to ensure that the assembly is not levitated by the flow during operation, but the force is optimised so as not to cause excessive distortion to the guide tubes.

254   The mixing of the coolant caused by the mixing vanes, has been deduced experimentally and accounted for in the analysis of the CHF. This approach is standard for the industry.

255   The likelihood of fuel cladding failure due to fretting has been addressed by changes to the assembly design. The AFA 3GLE twin bottom grid was first implemented in 2002 to improve clamping of the fuel rod in the grids after some cases of fuel rod fretting damage. The fix appears to have been successful and has been adopted. Significant reduction of wear was confirmed by CEA HERMES-P loop tests and experience feedback.

256   The key determining factor for the risk of fretting failures is the coolant flow velocity, so I have compared the UK EPR value with that of N4 plant. The flow rate in each assembly in the UK EPR design is expected to be significantly below that of N4 plants in which similar assemblies have been used. Furthermore, the design of the lower plenum is intended to lead to a more uniform inlet flow distribution. The likelihood of significant problems with fretting wear is expected to be remote and this will be proved by experience in the French EPR currently being constructed at Flamanville.

4.11.3   Findings

257   Overall, I did not find any significant shortcomings in the analysis of the fuel assembly structure.

4.12   Non-fuel Core Components

258   The principal non-fuel core components comprise:

- neutron source assemblies;

- RCCAs – also referred to as control rods; and

- thimble-plug assemblies.

259   The neutron sources consist of rods of radioactive material encapsulated in cladding tubes and suspended from a manifold structure designed to ensure that each rod assembly inserts into a guide tube within the fuel assembly. The rods are designed to provide sufficient neutrons to ensure that the reactivity of the core can be adequately monitored when in a shutdown state and that protection systems can function as required.

260   The RCCAs are similar except that they consist of material designed to absorb neutrons and are able to be raised and lowered within the guide tubes to control core reactivity and axial power shape. The intention is to operate the reactor with the RCCAs withdrawn from the core as far as practical - to achieve optimal fuel utilisation and to maximise their effectiveness in shutting down the reactor.

261   The thimble plugs, as the name suggests, are designed to plug the tops of the guide tubes to prevent the coolant flow bypassing the fuel. They allow just enough flow to prevent boiling inside the tubes.

4.12.1   EDF and AREVA's Case

262   These components are designed against thermal and structural limits to ensure that radioactive material is contained and that their continued functioning is ensured.

263   The design criteria against which non-fuel components are analysed are detailed in Chapter 4 of the PCSR and in Ref. 26. Mostly, they derive from AFCEN standards given in Ref. 18. The RCCA design is described in Ref. 61.

4.12.2   Assessment

264   I have assessed the design of these components and the design criteria applied by making comparisons with the components loaded at Sizewell B. The designs are very similar and the design constraints are similar.

265   The notable exception in this comparison is the use of boron carbide over part of the length of the RCCAs in place of the silver-indium-cadmium absorber material used at Sizewell B. Boron carbide is a standard product feature in rest of Europe, but is new to the UK. I asked for justification of this change.

266   The use of boron carbide increases the shutdown worth of the rods substantially and marginally reduces the burden of activated waste on disposal. Gas release from the boron is not a significant problem because the boron segment is generally out of the active core during power operation. These arguments provide a reasonable rationale. There is potentially some impact on the consequences of severe accidents. The removal of the silver inventory is a benefit radiologically, but the boron carbide has the potential to burn, which could affect the progression of the accident. My judgement was that these issues are not strong enough to balance against the benefits of increased rod worth and therefore I did not examine them in detail.

267 The considerable experience with these components to date does not indicate any significant concerns; although I am aware that flow-induced vibration can result in wear to the ends of the control rods. Experience at Sizewell B shows that this can be managed by routine inspection. Sizewell B also has HARMONI® RCCAs with both ion-nitrided stainless steel cladding and long solid tips and this significantly reduced wear rates experienced. This issue will need to be reviewed when as-built flow rates become apparent and reflected appropriately in the maintenance schedule.

4.12.3 Findings

268 Overall, I am satisfied that the design of incore components appears suitable for UK EPR.

> **AF-UKEPR-FD-06 -** *The licensee shall, before power raise, review as-built flow rates and reflect conclusions for flow-induced wear in the maintenance schedule for affected components.*

4.13 Long-term Storage of Spent Fuel in Interim Storage Facilities

269 The topic of long-term dry storage of fuel is the subject of significant current discussion as new facilities are designed. The issues are reviewed in Ref. 32. ND procured its own review of the subject for the UK EPR fuel and this is reported in Ref. 34.

270 The current plan is to store the fuel in the reactor pond until the heat generated by fission product decay has fallen sufficiently, then in the interim (between removal from the pond and final disposal) to load a number of assemblies into casks, filled with inert gas. This interim storage will be used long term until the fuel condition is suitable for final disposal.

271 SAP RW.5 requires that the safety case should identify the limits and conditions required for safe fuel storage. A number of the factors requiring consideration are significantly impacted by the prior operation of the fuel and therefore, while I recognise that the details of the proposed long-term fuel storage have not been finalised, I have required that fuel limits be defined to ensure that the design of these facilities can remain consistent with the proposed constraints on fuel operation.

4.13.1 EDF and AREVA's Case

272 Significant operating experience has been accumulated over a number of decades on both wet and dry spent fuel storage. The design objectives are:

- to prevent fuel cladding failure; and
- to preserve the fuel assembly structure to allow safe handling for retrieval operations.

273 Design criteria ensuring these objectives are detailed in Ref. 32. To date, more than 3 million M5™ fuel rods have been irradiated, with assembly and pin irradiations reaching 68 and 80 MWd/kgU respectively. Excellent oxidation kinetics is observed, with very low hydrogen take up compared to Zircaloy.

274 Post-irradiation examination shows little evidence of radial-orientated hydride precipitates (that could assist cracking) and no evidence of a hydride rim.

275 Degradation in wet storage is discounted due to the selection of corrosion resistant materials and to the low temperatures.

276   In dry storage, the temperature of the fuel is limited by design to ensure that thermal creep of the fuel cladding will not exceed the material ductility. It is accepted that hydrogen present in the cladding may dissolve as the fuel is heated on drying and subsequently may precipitate in a radial orientation. However, at expected levels of hydride, it is argued that the precipitates will not be sufficiently numerous to render the material brittle.

277   The case proposes bounding hydride levels of 200 ppm and pin internal pressures of 150 bar at 400C.

4.13.2   Assessment

278   I have considered the various failure mechanisms for spent fuel in long-term storage. Potential fuel degradation mechanisms include:

- clad strain resulting from rod internal pressure;

- corrosion;

- hydride embrittlement; and

- stress-corrosion cracking.

279   I have considered these topics in the context of the objective of retaining fuel cladding integrity. They are each discussed below.

280   I have not assessed irradiation damage of the fuel material during storage. I note that accelerated test programmes are providing detailed information on this topic, but Ref. 34 concluded that the cumulative self-irradiation damage does not destroy the crystallographic structure of the fuel within the time period envisaged for interim storage (100 years).

4.13.2.1 Cladding Creep

281   Clad strain is avoided by ensuring that the temperatures remain sufficiently low for creep rates to be small. Independent calculations with the ENIGMA fuel performance code have confirmed that the claims made are credible (Ref. 34).

282   I have requested information on fission-gas release as a result of alpha-particle decay. The analysis indicates that this is too slow a process to have a bearing on gas pressures during interim dry storage, which fall fairly rapidly in the early days due to reductions in decay heating.

4.13.2.2 Corrosion

283   Corrosion is avoided by chemistry control in wet storage and by an inert atmosphere in dry storage. The assessment of corrosion issues relating to fuel storage in the pond is addressed in the chemistry topic area.

4.13.2.3 Hydride Embrittlement

284   Hydride embrittlement is avoided by limiting the level of hydrogen uptake during irradiation and also by limiting the cladding hoop stress to levels where hydrogen-assisted cracking is unlikely.

285   The cladding ductility is potentially affected by any reorientation of hydride precipitates within the material, but the expected hydride levels are very low for M5™ cladding. Not only is the level of oxidation experienced in service low, but the hydrogen uptake resulting from this is substantially lower than Zircaloy at the same level of oxidation. Conventionally, the hydride limit for Zircaloy cladding has been set at 600 ppm, but M5™ cladding typically achieves levels below 100 ppm and the limit in Ref. 32 is set at 200 ppm. This is important because the proposed limit on clad stress during dry storage is relatively high and at the proposed level, some reorientation of hydride could be expected. The information presented in Ref. 32 brings me to the view that as hydride levels are reduced, the lower density of circumferential hydride precipitates makes the formation of radial hydrides more likely and it is quite possibly impractical to set a stress level where radial hydride can be entirely precluded.

286   EDF and AREVA argue in Ref. 36 that at these low levels of hydride, the cladding is likely to remain ductile. However, the argument is supported by limited evidence and will need to be confirmed by research programmes currently underway. The results of these programmes will need to be reviewed before core designs are finalised. I do not believe that this will lead to a serious problem because experience with dry storage of fuel has been essentially positive.

287   In addition to the effect of hydride precipitates in the bulk of the material, hydrogen potentially leads to Delayed Hydride Cracking (DHC). This is crack growth aided by brittle hydride precipitation at the crack tip. The phenomenon presumes the pre-existence of an incipient crack and requires significant mechanical loading.

288   The effect of hydrogen in the cladding on the potential for accelerated crack propagation has been observed in the zirconium alloy pressure tubes of the CANDU reactors. This has been extensively studied in the CANDU context (Refs. 37 and 39). While the CANDU alloy is not exactly the same as M5™, it also consists of zirconium with a dispersion of niobium precipitates.

289   The experimental results have revealed that the DHC phenomenon can be described by a dependence of the crack growth rate on the applied stress intensity at the crack tip (Ref. 36). These studies have shown that initiation of DHC occurs only if the stress intensity exceeds a threshold value, which appears to be approximately invariant.

290   The threshold value for the M5™ cladding is not directly available, but data is available for a range of Zirconium alloys and they behave in a consistent way. EDF and AREVA argue that this data is applicable. I lay particular weight on data for the E110 alloy which has a similar composition and grain size to M5™ and I judge this to be applicable.

291   Assuming an expected hoop stress under dry storage around 100 MPa, EDF and AREVA calculate that the depth of the incipient crack required to exceed the threshold stress intensity is greater than the cladding thickness and therefore can not be a cause of cladding failure (Ref. 36). This appears to discount DHC as a fracture mechanism for the cladding tubes.

292   I can not discount the possibility that cracking may occur within the welds of the fuel assembly skeleton, but the likelihood will depend upon the extent of defects within the welds and here I note that the assembly skeleton has a high degree of redundancy. I rely on the experience with spent fuel to date which does not indicate an operational problem.

4.13.2.4 Stress-corrosion Cracking

293   EDF and AREVA argue that fuel temperatures during storage are such that most mobile elements such as iodine or caesium will not be released at a sufficient rate to cause

stress-corrosion cracking (SCC) and that the stresses will be below the required threshold. Ref. 34 concludes that the risk of failure by SCC can be disregarded if the temperature remains below 420°C.

4.13.3 Findings

294   These arguments seem reasonable based on in-reactor experience and I am satisfied that it is likely to be possible to design a suitable storage facility for interim storage of spent UK EPR fuel subject to the reservation that more data is required on the effect of hydride precipitates.

> **AF-UKEPR-FD-04 -** *The licensee shall, before receipt of fuel on site, acquire and report data on hydride reorientation to demonstrate that irradiated cladding with predominantly radially-orientated hydride precipitates can retain adequate ductility at the hydride levels proposed.*

4.14   Fuel Performance in Reactivity Faults

295   The rapid insertion of reactivity into the core can, in certain circumstances, occur faster than the thermal response of the fuel pin. In these cases, the fuel is potentially subject to temperatures and stresses not encountered in other conditions.

296   The approach to assessment of the performance of fuel under these demanding conditions has been to combine data from a sequence of fuel-pin experiments with modelling of the power transient. Historically this has focused on avoiding fuel and cladding melt, but more recently, a more conservative approach has been adopted; to avoid extensive fragmentation and potential dispersal of high-burnup fuel by setting a fuel enthalpy limit at a level that would prevent cladding fracture. In RO 60, I requested further justification of the EDF and AREVA approach in the context of recent experiments.

4.14.1   EDF and AREVA's Case

297   A revised safety criterion, designed to avoid cladding failure, is proposed in Ref. 48. A Pellet-Cladding Mechanical Interaction (PCMI) cladding failure limit is defined which, when coupled with a DNB threshold, precludes most of cladding failures and meets the safety criteria requirements.

298   The PCMI failure limit is based on the notion of critical Strain Energy Density. The Strain Energy Density appears to be the most relevant parameter to take into account the mechanical state of the cladding. This failure limit is transposed to a fuel enthalpy and maximum enthalpy increase by means of code calculations.

299   The analysis uses the SCANAIR fuel pin code to model fuel pin transients. It has been qualified against suitable fuel pin power transient tests.

300   The final step of the analysis is a 3D simulation of the transient with the SCIENCE nuclear code package. The 3D core model uses bounding nuclear and thermal-hydraulic parameters to demonstrate that the design criteria for infrequent faults are met. The analysis method is detailed in Ref. 57. The limits on numbers of rods experiencing boiling crisis (DNB) and maximum enthalpy rise during the transient, are met.

4.14.2 Assessment

301   The criterion proposed for fuel enthalpy is supported by the available data, although I note that new nuclear-powered transient tests are planned in the CABRI reactor. These should be more prototypic that those currently used and need to be assessed when available. The proposed criterion of a critical strain energy density is not new, is plausible and has been widely discussed (e.g. Ref. 58).

302   The performance of the fuel pin in fast reactivity transients is represented in the SCANAIR computer code, which is used to develop a criterion for fuel failure based on fuel-pellet enthalpy. SCANAIR includes semi-empirical models of the main processes determining the fuel performance in these fast transients and has been qualified by comparison against experimental data. The code is widely used and therefore I focused my assessment on the modelling of high-burnup phenomena. Even under these relatively challenging conditions, I found the strain predictions made by SCANAIR surprisingly good and I echo the conclusions of Ref. 58, that the modelling of these phenomena is satisfactory, given the limited amount of prototypic data.

303   Having established a critical Strain Energy Density, a detailed response surface is developed which relates permitted enthalpy deposition to initial power level and burnup, based on the existing corrosion data for M5™ cladding. The excellent corrosion performance of the cladding to date results in a relaxed criterion, but I believe that this is justified provided that the oxide performs as expected.

304   The transient core response is based on a very bounding neutron kinetic calculation where limiting values of fuel Doppler, RCCA worth and delayed neutron fractions are assumed. I judge that this introduces satisfactory pessimism into the calculation.

305   The analysis demonstrates large margins to the PCMI failure criterion, and in most cases, demonstrates no fuel entering DNB despite a relatively large reactivity injection. However in the case initiated from approximately 40% power, the insertion limit is not sufficient to protect against DNB and about 2% of the fuel is predicted to overheat.

306   No analysis has been presented to demonstrate that the fuel entering DNB remains in a coolable geometry and it may be reasonably practical to change the rod insertion limit to prevent this. However, given the conservative nature of the analysis and the performance predicted for the bulk of the conditions, I am confident that EDF and AREVA will be able to provide satisfactory analysis to demonstrate that all limits are met.

4.14.3 Findings

> **AF-UKEPR-FD-07 -** *The licensee shall, before the RPV is installed, revise their reported analysis of the RIA fault to demonstrate that no fuel breaches the clad temperature limits designed to ensure residual ductility and provide an assessment of whether it may be reasonably practical to change the rod insertion limit to prevent any fuel entering the DNB condition.*

> **AF-UKEPR-FD-08 -** *EDF and AREVA shall, during the operational phase, review the derived criteria for cladding failure in RIA faults in the context of the results of the relevant experiments in the current CABRI programme if they become available.*

4.15   Fuel Performance in Loss of Coolant Accidents

307   The performance of the fuel and core is generally not sensitive to the detail of loss-of coolant accidents provided that the fuel remains covered by water. This is therefore generally the focus of the fault analysis and is assessed in Ref. 60.

308   In the event of a large LOCA, uncovery does occur, but the emergency core cooling system is designed to reflood the fuel before significant damage occurs. The assessment of this fault is covered in some detail in Ref. 60. Briefly, the analysis demonstrates that extensive fuel failures in this event are unlikely and hence that the core will remain coolable.

309   Some confirmatory calculations were made and these were in reasonable agreement with the EDF and AREVA analysis. This issue has therefore been satisfactorily addressed.

4.16   Accumulation of Slugs of Unborated Water

310   Formation of a slug of unborated water within the reactor cooling system, during shutdown conditions, represents a significant potential hazard because it is capable of causing a rapid reactivity transient should it be transferred unmixed to the reactor core. This has been recognised and measures intended to practically eliminate this risk from external sources of pure water have been taken. These measures are a combination of protection and administrative control. An adequate safety case justifying these measures has yet to be received and this will be raised as a GDA issue in Ref. 60.

4.17   Overseas Regulatory Interface

311   HSE's Strategy for working with overseas regulators is set out in (Ref. 72) and (Ref. 73). In accordance with this strategy, HSE collaborates with overseas regulators, both bilaterally and multinationally.

312   Interface with other regulators internationally has been provided principally by bilateral contact meetings with the US Nuclear Regulatory Commission and the Finnish regulator STUK. This helped me assign priorities to technical issues. The contacts were enabled through the Organisation for Economic Co-operation and Development's (OECD) Nuclear Energy Agency working group meetings in the context of the Multinational Design Evaluation Programme (MDEP).

313   In the case of a number of the fuel performance issues arising recently, work by the US NRC has informed the regulatory decision making. This is particularly true in the area of fuel performance in rapid reactivity faults and in high-temperature cladding oxidation where US NRC has taken a lead role in establishing a consensus.

314   STUK has provided useful intelligence across a wide range of issues.

315   The formal contact has been supplemented by attending an IAEA fuel expert meeting at Paul Scherrer Institute (PSI). Such meetings provide useful background information for judgements. The PSI meeting included a tour PSI research facilities and examination of PSI tests on dry storage of spent fuel.

4.18   Interface with Other UK Regulators

316   The fuel area interfaces with the Environment Agency indirectly in that fuel design and operation places demands on the design of facilities for long-term storage of the fuel and fuel design potentially influences radioactive discharges. However, direct contact within these areas has been made by my colleagues in the radioactive waste topic area.

4.19 Other Health and Safety Legislation

317   In assessing fuel and core design, my principle consideration has been to ensure that the fuel is constructed and operated in accordance with an appropriate safety case as required by the Health and Safety at Work etc. Act 1974 and its relevant statutory provisions. I have not considered other legislation.

5 CONCLUSIONS

318   This report presents the findings of the Step 4 Fuel and Core Design assessment of the EDF and AREVA UK EPR reactor.

319   To conclude, I am satisfied that the claims, arguments and evidence laid down within the PCSR and supporting documentation for the Fuel and Core Design are adequate. I consider that from a Fuel and Core Design view point, the EDF and AREVA UK EPR design is suitable for construction in the UK. However, this conclusion is subject to 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

320   The fuel design for the UK EPR, as detailed in the PCSR and supporting references listed in the Submission Master List (Ref. 77), is a development of existing fuel designs and the safety case documentation provides a rationale for the changes.

321   The fuel has been designed against an established set of criteria using conventional methods and the operational envelope is broadly consistent with that of existing fuel.

322   EDF and AREVA have responded to requests for consideration of reasonably practical safety enhancements by introducing additional operational constraints (for example the new RAPFE limit) and in the case of clad stress, they have engineered additional protection.

323   The fuel design has features which increase the margin to safety limits as the fuel reaches its limiting irradiation and the cladding material performs well, with low corrosion and hydrogen uptake. However, some additional data from ongoing research programmes will be needed for confirmation of the design constraints.

324   I conclude that the Assessment Findings listed in Annex 1 should be included in the programme for design and construction of this reactor as normal regulatory business.

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Table 1

Areas for Assessment During Step 4

Assessment Area	Description
Generic	Validation of computer codes and methodologies.
Nuclear Design	Review the claim that the moderator coefficient is always negative.
Nuclear Design	Ensure that controls that will be in place to ensure sufficient quantities of enriched boron are present.
Nuclear Design	Discuss with EDF and AREVA the requirements to meet: 1) the stuck rod criterion and; 2) ensure the fuel will be maintained sufficiently subcritical such that removal of a RCCA will not result in criticality.
Nuclear Design	The demands placed on the operator and the control system of control banks will need to be explored further in order to ensure that control and shutdown margin requirements are met.
Nuclear Design	Examine the potential for core misloading.
(IB & LBLOCA)	Independent assessment of the modelling of core reflood Clad ballooning and blockage.
Clad Stress	Assess revised case against PCI when available.
Fuel Irradiation	Assess evidence for high-burnup effects at an irradiation of 62 MWd/kgU.
Fuel Pin	Review the design substantiation against structural, thermal and Neutronic criteria.
M5 Performance	Examine data and arguments related to outliers in more detail.
CHF	The effect of crud and assembly bowing will be reviewed.
Fuel Assembly	Design changes to the structure will be reviewed.
RAPFE	Review justification for proposed limit.
Modelling	Review the adequacy of fuel modelling.
Design Criteria	A more detailed assessment of reactor core design criteria. Consideration of the adequacy of controls to ensure that the safety case boundary is intact. EDF and AREVA need to outline their proposals for continuous compliance with the Technical Specifications.
Crud	Review the fuel-performance aspects of the proposed chemistry strategy.
Clad Surface	Review the control of clad surface condition proposed.
Void fraction	Review the justification of the void fraction limit.
Long-term Fuel Storage	Review justification of the fuel limits in the context of EDF and AREVA's spent fuel storage plans.

Table 2

Relevant Safety Assessment Principles for Fuel and Core Design Considered During Step 4

SAP No.	SAP Title	Description
EKP - Engineering Key Principles		
EKP.1	Inherent safety	The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
EKP.2	Fault tolerance	The sensitivity of the facility to potential faults should be minimised.
EKP.3	Defence in depth	A nuclear facility should be so designed and operated that defence in depth against potentially significant faults or failures is achieved by the provision of several levels of protection.
ERL - Reliability Claims		
ERL.1	Form of claims	The reliability claimed for any structure, system or component important to safety should take into account its novelty, the experience relevant to its proposed environment, and the uncertainties in operating and fault conditions, physical data and design methods.
ERL.2	Measures to achieve reliability	The measures whereby the claimed reliability of systems and components will be achieved in practice should be stated.
EAD - Ageing and Degradation		
EAD.1	Safe working life	The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage.
EAD.2	Lifetime margins	Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components that are important to safety.
EMT - Maintenance, inspection and testing		
EMT.1	Identification of requirements	Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.
FA - Validity of Data and Methods		
FA.4	Fault tolerance	DBA should be carried out to provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures.

Table 2

Relevant Safety Assessment Principles for Fuel and Core Design Considered During Step 4

SAP No.	SAP Title	Description
FA.9	Further use of DBA	DBA should provide an input into the safety classification and the engineering requirements for systems, structures and components performing a safety function; the limits and conditions for safe operation; and the identification of requirements for operator actions.
FA.17	Theoretical models	Theoretical models should adequately represent the facility and site.
FA.18	Calculation methods	Calculational methods used for the analyses should adequately represent the physical and chemical processes taking place.
FA.19	Use of data	The data used in the analysis of safety-related aspects of plant performance should be shown to be valid
FA.20	Computer models	Computer models and datasets used in support of the analysis should be developed, maintained and applied in accordance with appropriate quality assurance procedures.
FA.21	Documentation	Documentation should be provided to facilitate review of the adequacy of the analytical models and data.
FA.22	Sensitivity studies	Studies should be carried out to determine the sensitivity of the fault analysis (and the conclusions drawn from it) to the assumptions made, the data used and the methods of calculation.
FA.23	Data collection	Data should be collected throughout the operating life of the facility to check or update the fault analysis.
ERC - Reactor Core		
ERC.1	Design and operation of reactors	The design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor.
ERC.2	Shutdown systems	At least two diverse systems should be provided for shutting down a civil reactor.
ERC.3	Stability in normal operation	The core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their specified range.
ERC.4	Monitoring of safety-related parameters	The core should be designed so that safety-related parameters and conditions can be monitored in all operational and design basis fault conditions and appropriate recovery actions taken in the event of adverse conditions being detected.

Annex 1

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

Fuel Core Design – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-FD-01	The licensee shall review the fuel assembly measurements taken from the first core offload at Flamanville and determine the impact that the data has on the safety justification of the proposed core management.	This is required before receipt of fuel on site.
AF-UKEPR-FD-02	The licensee shall review the results of available EPR physics testing and confirm uncertainty allowances in the safety case.	This is required before first fuel load.
AF-UKEPR-FD-03	The licensee shall demonstrate that the procedures proposed for loading the reactor core with fuel will ensure that an uncontrolled criticality is incredible or that all reasonably practical measures have been taken to prevent this.	This is required before first fuel load.
AF-UKEPR-FD-04	The licensee shall acquire and report data on hydride reorientation to demonstrate that irradiated cladding with predominantly radially-orientated hydride precipitates can retain adequate ductility at the hydride levels proposed.	This is required before receipt of fuel on site.
AF-UKEPR-FD-05	The licensee shall repeat the recent OECD benchmark studies on boiling flow in rod bundles and update the FLICA qualification documents.	This is required before by first fuel load.
AF-UKEPR-FD-06	The licensee shall review as-built flow rates and reflect conclusions for flow-induced wear in the maintenance schedule for affected components.	This is required before power raise
AF-UKEPR-FD-07	The licensee shall revise their reported analysis of the RIA fault to demonstrate that no fuel breaches the clad temperature limits designed to ensure residual ductility and provide an assessment of whether it may be reasonably practical to change the rod insertion limit to prevent any fuel entering the DNB condition.	This is required before the RPV is installed.

Annex 1

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

Fuel Core Design – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-FD-08	The licensee shall review the derived criteria for cladding failure in RIA faults in the context of the results of the relevant experiments in the current CABRI programme if they become available.	This is required During Operational phase.

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.

Annex 2

GDA Issues – Fuel and Core Design – UK EPR

There are no GDA Issues for this topic area.