New Reactors Division

Step 4 Assessment of Fuel & Core Design for the UK Advanced Boiling Water Reactor

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EXECUTIVE SUMMARY

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

This assessment report is my Step 4 assessment of the Hitachi-GE UK ABWR reactor design in the area of Fuel and Core Design.

The scope of the Step 4 assessment is to review the safety, security and environmental aspects of the UK ABWR in greater detail, by examining the evidence, supporting the claims and arguments made in the safety documentation, building on the assessments already carried out for Step 3 (Ref. 7). In addition I have provided a judgement on the adequacy of the Fuel and Core Design information contained within the Pre-Construction Safety Report (PCSR) and supporting documentation.

My assessment conclusions are:

- The GE14 fuel assembly design and the design of related core components are the result of an extended process of design improvements, which reduce the operating risk as far as reasonably practical;
- The Functional requirements of the design have been clearly defined and Design Criteria have been substantiated;
- Compliance with the Design Criteria has been demonstrated using qualified analysis methods and the basis for a suitable set of operating rules has been defined.

My judgement is based upon the following factors:

- Hitachi-GE report incremental improvement in the operational experience with their fuel, resulting in excellent fuel reliability;
- My visit to their manufacturing facility enabled me to witness their high levels of manufacturing quality;
- My sampling of the documentation supporting their analysis methods has satisfied me that the methods employed have a sound technical basis, and Hitachi-GE take appropriate account of uncertainty.

To conclude, I am satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for fuel design topic area. I consider that from a fuel design view point, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits beings secured.

The following matters remain, which are for a future licensee to consider and take forward in their site-specific safety submissions. They relate principally to design choices that remain to be finalised at later stages of the design or to decisions on the way the plant will be operated. These matters do not undermine the generic safety submission but require a licensee input/decision at a later date.

- In order to ensure that all reasonably practical measures are taken to reduce risk, the licensee shall at the appropriate time review the operating experience on debris-induced fuel failures to determine whether further mitigation measures are reasonably practical.
- To ensure that the increased risk associated with these operations is adequately managed, the licensee shall ensure that procedures for carrying out rod swap operation are performed at suitable reactor power levels to maintain adequate safety margin to accommodate frequent fault sequences without damage to the fuel cladding by stress-corrosion cracking.

- To establish an adequate set of analyses to support the safety case for the established dry storage design, the licensee shall substantiate that the fuel assembly dwell time in the spent fuel pool (required prior to transfer to interim storage) is consistent with the design requirements of the selected storage cask and associated systems. In particular, the qualification of this analysis will need to demonstrate that the complex physical processes taking place in the cask are adequately represented in the modelling.

- To reduce the reliance on the Core Monitoring System to levels commensurate with its safety classification, the Licensee shall design and implement an adequate diverse surveillance method to allow the operator to verify that the Core Monitoring System is performing correctly and that the plant is demonstrably within operating limits.

- To ensure that the level of surveillance given to the condition of the discharged fuel is appropriate to the uncertainty associated with the likely performance of the refuelling strategy the licensee choses, the licensee shall establish a policy on post-irradiation inspection, including inspection of failed fuel and take account of this in the design of fuel handling equipment.

- To provide adequate contingency arrangements for foreseeable fuel damage, the licensee shall review the technology available for the inspection and storage of damaged fuel and provide sufficient equipment and arrangements to ensure that suitable ALARP measures can be taken to meet the requirements for the storage of damaged fuel.

LIST OF ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
APRM	Average Power-range Monitoring
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing of Materials
BMS	ONR Business Management System
BSL	Basic Safety level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
BWR	Boiling Water Reactor
CHF	Critical Heat Flux
CMS	core monitoring system
CPR	Critical Power Ratio
DHC	delayed-hydride cracking
EA	The Environment Agency
FCI	Fuel-Coolant Interaction
FMCRD	Fine-Motion Control Rod Drive Mechanism
GDA	Generic Design Assessment
GNF	Global Nuclear Fuels Ltd.
GRS	Gesellschaft Fuer Reaktor Sicherheit
IAEA	The International Atomic Energy Agency
INPO	International Nuclear Power Operators
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-coolant Accident
LPRM	Local Power-range Monitoring
LWR	Light Water Reactor
MCPR	Minimum Critical Power Ratio
MDEP	Multi-national Design Evaluation Programme
NCB	Non Classified Building
NDA	Nuclear Decommissioning Authority
OCNS	Office for Civil Nuclear Security
OECD	Organisation for Economic Cooperation and Development
OJEU	Official Journal of the European Union
ONR	Office for Nuclear Regulation
PCER	Pre-construction Environment Report
PCI	Pellet-cladding Interaction
PCMI	Pellet-cladding Mechanical Interaction

PCSR	Pre-construction Safety Report
PID	Project Initiation Document
PMS	Plant Management System
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
PWR	Pressurised Water Reactor
RAPFE	Radial-averaged Peak Fuel Enthalpy
RGP	Relevant Good Practice
RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPS	Reactor Protection System
SAPs	Safety Assessment Principles
SCC	Stress Corrosion Cracking
SCRRI	Selected Control Blade (Rod) Run-In
SFAIRP	So Far As Is Reasonably Practicable
SFP	Spent Fuel Pond
SFIS	Spent-fuel Interim Store
SSC	System, Structure and Component
SSER	Safety, Security and Environmental Report
STUK	The Finish Nuclear Safety Authority
TAG	(Nuclear Directorate) Technical Assessment Guide
TIP	Traversing In-core Probe
TMOL	Thermal-Mechanical Operating Limit
TQ	Technical Query
TSC	Technical Support Contractor
US NRC	Nuclear Regulatory Commission (United States of America)
WENRA	The Western European Nuclear Regulators' Association

TABLE OF CONTENTS

Annex(es)

1 INTRODUCTION

1. This assessment report details my Step 4 Generic Design Assessment (GDA) of Hitachi-GE's UK ABWR reactor design in the area of Fuel Design.

1.1 Background

2. Information on the GDA process is provided in a series of documents published on our website (**http://www.onr.org.uk/new-reactors/index.htm**). The outcome from the GDA process sought by Requesting Parties such as Hitachi-GE is a Design Acceptance Confirmation (DAC) issued by ONR and a Statement of Design Acceptability (SoDA) issued by the Environment Agency (EA) and Natural Resources Wales (NRW).

3. The GDA of the UK ABWR has followed a step-wise approach in a claims-arguments-evidence hierarchy which commenced in 2013. Major technical interactions started in Step 2 with an examination of the main claims made by Hitachi-GE for the UK ABWR. In Step 3, the arguments which underpin those claims were examined. The reports in individual technical areas and accompanying summary reports are also published on ONR's website.

4. The objective of the Step 4 assessments is to undertake an in-depth assessment of the safety, security and environmental evidence. Through the review of information provided to ONR, the Step 4 process should confirm that Hitachi-GE:

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

5. During Step 4 I have undertaken a detailed assessment, on a sampling basis, of the safety and security case evidence. The full range of items that might form part of the assessment is provided in ONR's GDA Guidance to Requesting Parties (http://www.onr.org.uk/new-reactors/ngn03.pdf). These include:

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

6. The regulatory observations (ROs) issued to Hitachi-GE as part of my assessment are also published on our website, together with the corresponding Hitachi-GE resolution plan.

1.2 Scope

7. The scope of my assessment is detailed in an assessment plan (Ref. 8). This focused on the justification for design criteria and the adequacy of the methods of assessment used to confirm safety margins; including the validation of the methods and the treatment of uncertainty in their qualification. Often the application of these methods is assessed in another topic area (for example Fault Analysis).

8. In the case of the interim dry storage, Hitachi-GE presented a preliminary design of fuel storage cask and therefore my assessment limited itself to consideration of the design criteria against which the detailed design will be analysed. The thermal analysis of the Holtec Cask Design has not been assessed. This is appropriate for GDA because there are a number of potential cask designs and the selection of the most appropriate design will be made by the licensee.

1.3 Method

9. The methodology for the assessment follows HOW2 guidance on mechanics of assessment within the Office for Nuclear Regulation (ONR) (Ref. 1).

10. This assessment has been focussed primarily on the definition of fuel and core functional requirements and the identification of degradation mechanisms which could impair the performance of the system. Having identified these physical processes, I sampled the justification of relevant fuel design criteria and the setting of operational limits and protection levels to ensure that the criteria are met. I confirmed that the claims made are supported by suitable evidence and that the treatment of uncertainty is appropriate to the level of risk and the magnitude of the margin to safety limits.

2 ASSESSMENT STRATEGY

11. ONR's GDA Guidance to Requesting Parties (http://www.onr.org.uk/new-reactors/ngn03.pdf) states that the information required for GDA may be in the form of a Preconstruction Safety Report (PCSR). At a high level, Ref. 78 advises judgement on the adequacy of the nuclear safety case. Adequacy is based on the following expectations:

- the legal requirement that the level of risk will be reduced ALARP;
- national and international relevant good practice is incorporated into the design;
- the evolution of previous designs has led to improved safety;
- risk assessment has been used to identify potential improvements; and
- risks achieve our basic safety objectives if reasonably practicable.

12. Technical Assessment Guide (TAG) 051 sets out regulatory expectations for a PCSR (http://www.onr.org.uk/operational/tech_asst_guides/ns-tast-gd-051.pdf).

2.1 Standards and Criteria

13. The standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAPs) (Ref. 2), internal TAGs (Ref. 3), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites.

Safety Assessment Principles

14. The key SAPs applied within the assessment are presented in Annex 1.

Technical Assessment Guides

15. The TAGs that have been used as part of this assessment are set out in Annex 2.

National, International Standards and Guidance

16. I informed my expectations by considering a number of standards provided by international bodies. The principal sources are IAEA and WENRA documents as detailed below.

- IAEA standards (Ref. 5) include general safety principles and specific safety requirements as follows, but I also judge that specific examples of good practice relating to reactor core design are found in the relevant safety guide:

  Fundamental Safety Principles. IAEA Safety Standards Series, SF-1, IAEA Vienna, 2006;

  Safety Assessment for Facilities and Activities, General Safety Requirements No. GSR Part 4 Rev 1, IAEA Vienna, 2016;

  Safety of Nuclear Power Plants: Design Safety Standard - Specific Safety Requirements SSR 2/1 IAEA 2016; and

  Design of the Reactor Core for Nuclear Power Plant, IAEA Safety Guide, NS-G-1.12, IAEA, Vienna 2005.

- WENRA references (Ref. 4):

  Reactor Safety Levels for Existing Reactors (September 2014);

  Safety Objectives for New Power Reactors (December 2009);

  Statement on Safety Objectives for New Nuclear Power Plants (November 2010); and

  Waste and Spent Fuel Storage Safety Reference Levels (February 2011).

17. I also considered specific examples of practices relevant to particular issues. These were found in IAEA and NEA topical reports and are referenced from the body of my assessment as appropriate.

2.2 Use of Technical Support Contractors (TSCs)

18. It is usual in GDA for ONR to use technical support contractors, for example to provide additional capacity, to enable access to analysis techniques and models, and to enable ONR‘s inspectors to focus on regulatory decision making.

19. The justification of the fault tolerance of a reactor core is fundamentally dependant on the adequacy of the calculations predicting core neutronic behaviour and the associated reactor power and fuel temperatures. Initially, I had some concerns about the accuracy claims made for the reactor core models used by Hitachi-GE (given their relative age). I therefore commissioned an independent core model with Gesellschaft Fuer Reaktor Sicherheit (GRS). This allowed me to compare the results of sample calculations from Hitachi-GE codes with those of diverse analysis methods and confirm Hitachi-GE’s claims.

2.3 Integration with other assessment topics

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

21. Reactor Chemistry provides input to the assessment of crud, activation and corrosion aspects of the Fuel Design assessment. We held cross-cutting meetings to assess the water chemistry limits for fuel design.

22. The Fuel Design assessment provides input to the core performance and design criteria aspects of the fault studies assessment. The assessment of faults is led by fault studies. I provided assessment of the adequacy of the models used in some fault studies and in the case of the core model, the quantification of uncertainty in the fuel thermal hydraulic model. I examined the relevant models and correlations in the TRACG code and fed this information into the assessment reported in that topic report.

23. I also provided advice on the modelling of fuel in fault transients and severe accidents. In particular, the adequacy of the JASMINE modelling of molten core coolant interaction; which is reported here.

24. In the area of spent-fuel management, I provided advice to the relevant inspectors on the adequacy of fuel performance modelling used in the analysis of the fuel pool and fuel route and I sampled the design criteria to be used for spent-fuel storage analysis.

25. In addition to the above, there have been interactions between Fuel Design and the rest of the technical areas. These interactions have been two-way, and mostly, of an informal nature.

2.4 Sampling Strategy

26. 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.

27. The sampling strategy for this assessment was to identify areas where the approach was novel to the UK, the hazard significant or the uncertainty comparable to available safety margins and to examine the arguments and evidence supporting the claim that risks have been reduced as far as reasonably practicable.

3 REQUESTING PARTY'S SAFETY CASE

28. The Safety case in the fuel and core topic area is set out for Step 4 in the PCSR Chapter 11: Reactor Core (Ref.40) and Chapter 19 Fuel Storage and Handling (Ref.62). The principle claims made are summarised below:

29. The fuel mechanical design is detailed in Ref. 20. It comprises of a 10x10 fuel pin array that includes full length fuel pins, part length fuel pins and 2 large central water rods. The cast stainless steel lower tie plate includes a conical section that seats into the fuel support and a grid that establishes the proper fuel pin spacing at the bottom of the bundle. The lower tie plate also houses a debris filter to prevent debris from entering the assembly; potentially leading to fretting failure of the fuel pin cladding.

30. The top of the bundle includes an upper tie plate which as well as locating the top of the fuel pins, provides the handle that is used to lift the bundle for transferring the fuel bundle from one location to another.

31. Another key part of the bundle is the set of 8 spacer grids, which functions to maintain proper spacing between the fuel pins along the axial length of the bundle as well as influencing critical power performance.

32. Fuel channels for the GE14 design have thinner sides and thicker corners to provide sufficient strength in the regions of highest stress while minimizing material for improved neutron economy.

33. In the UK ABWR, the fission rate is reduced by an increase in the steam void content in the moderator. Therefore any system input that increases the reactor power, either in a local or global sense, produces an additional steam void that reduces the reactivity and therefore the power. This void feedback effect is one of the inherent safety features of the UK ABWR system.

34. The GE14 assembly is designed with two large circular water rods that occupy eight fuel pin lattice positions. They affect the fuel assembly neutronic performance and provide support for the spacer grids.

35. Expansion springs provide sufficient axial spacing for the differential growth of the full length fuel pins and water rods.

3.1 Operational Experience

36. In recent years, the fuel produced by Global Nuclear Fuels Ltd (GNF) has experienced no fuel failure events due to mechanical design problems or component failures, and no failures due to crud, corrosion, or flow induced vibration. Most fuel failure events (almost 80 percent) have been due to debris fretting, but the introduction of an improved debris filter lower tie plate has resulted in a reduction in failures from debris by approximately a factor of 5 in GNF 10×10 fuel. The only other significant failure mechanism observed has been a total of [11] PCI-type, "duty-related" failures, in a small number of reactor manoeuvres (most more than 10 years ago). The likelihood of this type of failure has been reduced through several measures including improved operating guidelines and improved pellet quality.

37. The reference fuel design in the UK ABWR GDA (designated as GE14) has been deployed in reload quantities for over 15 years. The experience base exceeds 3 million fuel pins.

3.2 Functional Requirements of the Fuel

38. The fuel has the following functional requirement in normal operation and frequent design basis faults:

- Containment of radioactive materials;
- Reactivity control and safe core shutdown; and
- Preservation of geometry for heat removal.

39. This is achieved in the design by respecting a set of prescribed design criteria; designed to prevent the following failure mechanisms from acting:

- Excessive cladding creep out rate due to fuel pin internal pressure;
- Fuel melting;
- Cladding fatigue and excessive stress generally;
- The fuel cladding damage from collapse into a fuel column axial gap;
- Localized reduction in cladding ductility due to oxidation and hydriding;
- Significant vibration and consequent fretting wear, particularly at spacer–fuel pin contact points; and
- PCI failure that could occur due to a combination of tensile stress from Pellet Cladding Mechanical Interaction (PCMI) and corrosion from fission products, such as iodine.

40. The functional requirements of the nuclear design are that reactivity is controlled to enable the chain reaction to be stopped under all circumstances and the reactor returned to a safe shutdown state.

41. The power distribution is controlled so that the operational thermal limits (Maximum Linear Heat Generation Rate and Minimum Critical Power Ratio) are not exceeded in normal operation and all frequent design basis faults (except where specific justification is made).

42. Core designs are selected to satisfy the following conditions:

- The overall moderator void coefficient is negative.
- The Doppler coefficient is negative and has sufficient reactivity feedback characteristics to terminate infrequent design basis faults.
- The moderator temperature coefficient is not a value that may pose a safety concern in the current design range.
- The power reactivity coefficient is negative and its feedback is large enough to reduce the spatial oscillation of xenon.
- The reactor shutdown system is able to bring the core to a subcritical state, with adequate margin that bounds inherent biases and uncertainties associated with calculation methods and operating conditions.

43. Core performance parameters, such as power distribution, flow, thermal margins and exposure, are monitored so that the fuel integrity is maintained during the reactor operation.

3.3 Functional Requirements of the Control Rods

44. Diverse systems are provided for shutting down the reactor comprising control rods with multiple means of insertion and a backup provided by a fluid system; acting over a slower time scale.

45. Sufficient shutdown worth is provided to ensure that the core remains subcritical with the highest-worth pair of control rods removed from the core.

3.4 Design criteria

46. A set of design limits are defined and applied, in the fuel pin thermal-mechanical design analyses, to ensure that fuel pin mechanical integrity is maintained throughout the fuel pin design lifetime. The design criteria were developed to focus on the parameters most significant to fuel performance and on operating occurrences that can realistically limit fuel performance.

3.5 Limits and Conditions for Operation

47. Ref. 40 and its supporting documents identify a number of constraints that must be applied to ensure safe operation during normal, fault and accident conditions. Some of these constraints are maximum or minimum limits on the values of the plant parameters (e.g. pressure or temperature) whilst others prohibit certain operational states or require a minimum level of availability of a specified equipment. Those constraints are collectively defined in as Assumptions, Limits and Conditions for Operation (LCOs). These form the basis of the analysis to demonstrate the satisfactory performance of the fuel and will be the basis of operating rules as proposed in Refs. 43 and 52.

4 ONR ASSESSMENT

48. This assessment has been carried out in accordance with HOW2 guide NS-PER-GD-014, “Purpose and Scope of Permissioning” (Ref. 1) consistent with internal guidance on the mechanics of assessment within ONR (Ref. 6).

4.1 Scope of Assessment Undertaken

49. I derive my assessment expectations principally from ONR SAPs and TAG 75. I expect a safety case for fuel and core to demonstrate an understanding of the degradation mechanisms which potentially impair the performance of fuel and core components and their ability to deliver their safety functions.

50. In the case of the fuel, the principal requirement is confinement of nuclear material and in the case of the control rod, shutdown of the reactor is paramount (although other functional requirements are also important). I have examined the design of the fuel and core components to confirm that:

- the important physical processes which need to be accounted for are represented in the analysis and that
- where significant degradation mechanisms have been identified, reasonably practical measures have been taken to mitigate them.

51. I expect the Safety Case to present a set of design criteria which need to be respected in normal operation and fault conditions and these design criteria need to be substantiated by sufficient evidence in accordance with SAPs key principles.

52. Compliance with the design criteria is often ensured by setting suitable operating limits based on modelling and analysis. I expect the models to adequately represent the plant and the relevant physical processes. This needs to be validated by suitable testing (SAP AV.1-8).

53. In the event that the operating limits are breached, I expect safety systems to have sufficient redundancy and diversity to prevent the plant exceeding design criteria.

54. Analysis of these systems needs to demonstrate that there is high confidence of respecting the design criteria (SAP FA 1-11). I expect that analysis codes and methods used to substantiate the design are based on sound physical principles and are adequately qualified.

4.2 Justification of the Fuel Assembly Mechanical Design

55. My expectations for this topic are that: the design of fuel and core components will include consideration of potential loads that may be experienced in normal operation and fault conditions; the basis of the component design life will be established; and that maintenance and inspection regimes will be set to ensure that there is suitable margin to these limits in operation as advised by SAP ERL.1 and 2. This analysis needs to account for the effects of aging and degradation as appropriate.

Requesting Party's Case

56. The mechanical design of the fuel assembly is described in Ref. 20. The analysis systematically considers potential degradation mechanisms including: Static and dynamic loads, fatigue, creep and wear.

57. The design objectives are achieved by the imposition of mechanistic limits on the predicted performance of the fuel under the conditions of authorized operation. This analysis methodology is comprised of three elements:

    1. Design Criteria - Mechanistic design criteria are applied to parameters that realistically represent fuel integrity limitations.

    2. Analytical Fuel Performance Modelling, validated against extensive experimental measurements, enables quantification of the fuel performance.

    3. Statistical and worst tolerance analysis to allow a conservative quantification of the acceptability of a worst tolerance condition.

58. The analysis uses a combination of detailed load and stress analysis against established codes, experimental and in-service measurement and operational experience.

59. Hitachi-GE has discussed the development of their fuel design in Chapter 11 of the PCSR. It details a number of modifications to the fuel design to increase its fault tolerance and reduce the rates of in-service cladding failure.

60. A systematic programme of safety improvements has resulted in essentially eliminating most fuel cladding failure mechanisms. Selected items of note are detailed below:

Fuel pin Design

61. Chamfered pellet edges have reduced the likelihood that pellets will be damaged during the manufacturing process and inspection criteria tightened to help prevent damaged pellets leading to cladding failure in reactor (Ref. 20).

62. GNF also introduced a cladding inner liner of pure zirconium for protection against pellet-cladding interaction (PCI). The liner serves as a buffer between the Zircaloy-2 part of the tube and the uranium dioxide pellet (Ref. 17).

63. GNF work to a tighter iron specification; toward the high end of ASTM limits for Zircaloy-2. The cladding utilizes an optimal nodular corrosion resistant microstructure and has achieved an experience base of well over several million pins; with no corrosion or shadow corrosion related failures.

64. Hitachi-GE claims that the 10x10 pin bundle configuration reduces the Maximum Linear Heat Generation rate; achieving a substantial increase in performance and reliability.

Fuel Assembly Design

65. The current fuel assembly channel material is susceptible to bow, especially at high exposure. Channel bow influences fuel operating safety margins by perturbing the neutron spectrum, and potentially results in difficulties in inserting control rods; leading to early discharge of the fuel.

66. Hitachi-GE has responded to this problem by making explicit allowance for the effects of bow on safety margins and increasing the gap between fuel assemblies.

67. Having reduced the fuel pin failure rate to exceptionally low levels, the predominant failure mode in recent years has been the puncture of the cladding by flow-induced fretting; caused by debris trapped in spacer grids. Hitachi-GE has introduced the "Defender^{TM}" filter specifically targeted against swarf and wire-like debris. This further reduced the size of debris likely to enter the fuel. Operational experience shows that this modification can significantly reduce the incidence of fretting failures (Ref 51).

68. In the case of Fretting Wear, testing is performed to assure that the mechanical features of the design do not result in significant vibration and consequent fretting wear. The vibration response of the then new design was compared to a design that has demonstrated satisfactory performance through discharge exposure. Specifically the GE14 design vibration response was compared to the reference GE12 design for this purpose. The outcome was positive.

ONR Assessment

69. The design of the fuel assembly is substantiated by a mix of: relatively simple calculations compared against simple stress limits; more complex finite element models; and experimental measurements and demonstrations. I take comfort from the fact that Hitachi-GE claim that the assembly is compliant with the American Society of Mechanical Engineers (ASME) standard for fuel assembly design and the US NRC standard review plan (Ref. 20).

70. The finite-element modelling uses a well-established ANSYS code and therefore I chose not to consider code qualification. The analysis has been compared to the allowable stress for the component, for a conservative loading condition. In certain cases (for example the top nozzle) this has been supplemented by testing.

71. The testing appears to have been carried out for the more risk-significant components or for areas of technical uncertainty (such as the seismic loading and the fretting behaviour of spacer grids). I regard the approach adopted as pragmatic and reasonable.

72. Hitachi-GE proposes to irradiate the fuel to a peak pellet limit of [70 MWd/kgU], which is high (but within the current practice in Japan). I therefore asked Hitachi-GE to consider what irradiation limit was reasonably practical as a constraint on core designs. They advised that GE14 assembly design is routinely exposed to higher the fuel pin irradiation in the USA than in Japan and the UK safety case was constrained by the availability of data to qualify the fuel performance code PRIME at that time (Ref. 83). My assessment of the PRIME model is considered in more detail later in this report, but I consider their reasoning acceptable.

Measures Taken to Reduce Risk

73. The safety case for fuel design is principally constructed of claims relating to fuel degradation and core component degradation mechanisms and the suitability of operating rules designed to preserve the functional capability of the systems. However, UK law requires that measures are taken to reduce risk to employees and the public as far as reasonably practical. Hitachi-GE has addressed this issue by detailing certain design enhancements undertaken.

74. Hitachi-GE and GNF have effectively implemented a systematic improvement programme. As a result of this, the design is now highly optimised. The fuel failure rate observed in recent years is commendable. However some of the modifications currently under development are not yet implemented in the UK ABWR design and ONR will engage with prospective licensees to examine the lessons learned from experience in the further stages of this project.

Fuel pin Design

75. Various studies have confirmed the importance of avoiding pellet chips and I was pleasantly surprised to see how defect-free the pellets on the GNF production line in Wilmington (Ref. 9). The defect rates for GE14 fuel generally are well within benchmark expectations in TAG 75.

76. I have examined the effect of the cladding liner material; both in Hitachi-GE ramp-test data and in data from the Studsvik clad integrity programme (Ref. 49 and 86). This modification is helpful as a mitigation of the effect of pellet chips, but also has a wider benefit. It is worth about [15 KW/m] in safety margin for PCI compared to conventional cladding and provides sufficient safety margin [20%] between the limiting conditions of operation and the cladding design criteria. This has proved useful in making a fault-tolerance case against PCI stress-corrosion cracking.

77. The liner material also helps increase cladding resilience in storage. It is more able to absorb hydrogen. This is important during dry storage: As the hydrogen reaches its solubility limit in the adjacent material, rather than precipitate as hydride platelets, it migrates to the liner. This helps mitigate the potentially embrittling effect of hydrogen uptake (see Section 4.9). The cladding liner is a good example of the improvement work carried out and is important to giving me confidence in the fuel design.

Fuel Assembly Design

78. I am aware that channel bow is a safety and performance issue for BWR generally and has become an increasing problem in recent years as more aggressive core designs have led to fresh fuel operating adjacent to inserted control rods; causing galvanic shadow corrosion on one side of the channel box and hydride-induced growth (Ref.50). I therefore asked Hitachi-GE what measures it was taking to address the issue.

79. Hitachi-GE advised that channel distortion is minimised in the core design process by constraining the loading of fresh assemblies. If during operation, it is determined there is an increased susceptibility of control-rod interference in a given cell, Hitachi-GE provide guidance on how to monitor the cell to ensure the control rod drive is operable (Ref. 84).

80. In addition to monitoring channel box performance and periodically testing control-rod movement, the solution the industry has pursued is to change the channel-box material to make it more resistant to this type of corrosion. Hitachi-GE have a new box design

but they considered that there is not yet sufficient operating experience to offer this design for GDA (Ref. 61). I agree with this view. I expect that a potential licensee will review the position when more data is available. I judge that this project has sufficient momentum that I do not consider an assessment finding is necessary to mandate this. This is part of the normal process of fuel development.

81. From a debris fretting perspective, I am encouraged by the improved performance achieved with the debris filter in the lower tie plate. This is an area which will merit further review as more data becomes available. A feed water strainer is a further prospective countermeasure to reduce the occurrence of debris-induced fuel failures. Hitachi-GE has recommended that this be evaluated at a later stage by making use of operational experience gathered in the interim period. I expect that a prospective licensee will do this at the appropriate time during construction. To prompt a review, I judge that the following assessment finding is appropriate:

> In order to ensure that all reasonably practical measures are taken to reduce risk, the licensee shall at the appropriate time review the operating experience on debris-induced fuel failures to determine whether further mitigation measures are reasonably practical.

4.3 Core Design

82. The placement of fuel assemblies within the core is a significant determining factor for the core performance. I expect this to be the result of a systematic process which ensures that the core has optimal margins to safety limits throughout a cycle of fuel irradiation.

Requesting Party's Case

83. Hitachi-GE's core design studies for an equilibrium core are detailed in Ref. 97. Analysis of the initial core loading is found in Ref. 96. These studies are based on an 18 month refuelling interval and include:

- thermal margins – Linear Heat Generation Ratio (LHGR) and Minimal Critical Power Ratio (MCPR);
- reactivity margins (hot excess and cold shutdown margin; and
- reactivity coefficient evaluation.

84. The pins along the peripheral row of the bundle have reduced enrichment in order to mitigate power peaking due to high thermal neutron flux and to make allowance for potential channel bow.

85. Hitachi-GE demonstrates that under normal operation, the reactor is able to be operated without exceeding the thermal limits. The local power peaking is taken into account by deriving radial power factor (R-factor) values.

86. The start-of-cycle linear rating is high, but there is margin to the design limit. In the first days of operation the linear rating falls rapidly; maintaining a more comfortable margin for the rest of the cycle. An Operating Limit MCPR (OLMCPR) of [1.35] is predicted; the core design has been developed to maintain a 7% margin or greater to the OLMCPR.

87. Calculations have been performed to determine the effects of a mislocated or a misoriented fuel assembly in the core on the Critical Power Ratio (CPR). The effect is accommodated within the available safety margins.

88. Excess reactivity at-power is the natural reactivity of the core under design conditions which has to be accommodated by control rod insertion. This is designed to be

relatively constant (and modest) throughout most of the cycle, thereby minimising control blade adjustment to compensate for burn-up.

89. The calculations demonstrate that the core can be shutdown safely with a ΔK margin of 1%; including allowance for the worst single failure in the control rod system. The design avoids placing fresh fuel in control cells, which is good from an assembly bow perspective.

90. Analysis considers the adequacy of the core dynamic response and provides constraints required on the control rod movement.

ONR Assessment

91. The topics covered in the core design reports meet my expectations and demonstrate that it is possible to achieve a core design which is likely to be able to respect the safety limits identified in the fault analysis.

92. For the core, the initial power distribution is controlled by careful use of fixed burnable poison. The initial rod insertion is modest and the initial power shape is controlled partly by a higher-than-normal initial flow. Reasonable margins to safety limits are maintained.

93. The equilibrium core achieves a reasonable margin to fuel limits with a simple fuel utilisation strategy. The fuel burnup achieved is higher than I expected.

94. Overall, I am impressed with the skill used to optimise these core designs and I am content that the designs have met a reasonable set of requirements.

4.4 Control Rod Design

95. The control rods have important safety functions in limiting reactor power levels and shutting down the reactor, but they are also a source of active nuclear material which needs to be contained. I therefore considered the adequacy of the design limits proposed.

Requesting Party's Case

96. A description of the control rod is found in Ref. 90. Hitachi-GE argues that the control rod system has two functions:

1 Core Reactivity Control / Power Distribution

> The control rod provides the functionality of controlling core reactivity and adjusting the core power distribution. Core reactivity control is implemented by varying the control rod position, and the power distribution is adjusted by changing the locations of inserted control rods.

2 Shutdown

> The control rod provides the functionality of shutdown by inserting all of the control rods (both boron carbide type and hafnium flat-tube type) simultaneously.

97. The nuclear lifetime of the control rod is defined as [10%] reduction in relative reactivity worth over any quarter axial segment of the active length. On the other hand, the mechanical lifetime of the control rod is defined as the limit of structural integrity of the part which retains the neutron absorbing material.

98. Two types of control rod are used in UK ABWR:

- Boron carbide control rods are used for shutdown and remain out of the core during normal irradiation. These have high reaction cross sections and will be exchanged in typically [30] years.
- Hafnium rods retain their worth over a sequence of absorptions and are therefore preferred for control purposes; but have a lower initial reactivity worth.

Boron Carbide Rods

99. For UK ABWR, Hitachi-GE proposed that the boron carbide depletion is limited to [25%] to avoid tube failures and boron carbide washout.

Hafnium flat-tube type Control Blades

100. Hafnium has a long chain of absorption reactions, without losing much worth and therefore this type of rod is routinely inserted into the active core for control.

101. Despite their favourable nuclear properties, the hafnium Control Blades' life is often limited by nuclear depletion. Its life time with continuous irradiation is estimated as [three] years due to reaching the end of their nuclear life ([10%] loss of rod worth).

ONR Assessment

102. I consider that limits on control rod dwell are important, not only to ensure that the control rods retain their ability to provide their primary safety function of ensuring reactor shutdown, but also to ensure confinement of the nuclear material within the rods.

103. This is particularly the case for boron carbide rods, where irradiation-induced helium generation can lead to swelling of the material and potentially cracking of the cladding. Release of boron carbide into the coolant is undesirable because it impedes control of reactivity, but also because activation products such as tritium and radiocarbon isotopes present a small (but significant) radiological hazard. I therefore issued regulatory queries asking Hitachi-GE to provide justification of the limits and conditions of operation for the rods.

104. The design of the control rods is governed by a number of specific Japanese nuclear requirements, codes and standards. There are three codes:

    - Basic requirements specification (Ref. 91);
    - Allowable stress limits (JSME S NC1-2005/2007) - similar to ASME 3; and
    - Seismic requirements - met by testing (JEAC 4601-2008).

105. I have not required translation of these documents, but I have confirmed that they have been developed by a public consultation process, including consultation with the US NRC. The allowable stress levels have been found to be consistent with practice elsewhere.

106. The primary mode of failure is stress-corrosion cracking and this is addressed by the use of materials that have suitable levels of resistance, but sensitivity cannot be completely avoided; particularly in the region of welds. One aspect of Hitachi-GE's case is therefore defect tolerance.

Boron Carbide Rods

107. The limit on boron carbide rod irradiation is generally determined by the control rod dwell time in reactor. This can be determined by the rate of swelling of the absorber material. The intention is to prevent stress-corrosion cracking of the control rod cladding; leading to release of carbide into the coolant (Ref. 106). This cracking has occurred in a number of cases due to the use of rods for power suppression (Generally, the boron rods are fully withdrawn during power operation and their nuclear life is not an active constraint). In view of the potential for loss of confinement of nuclear material, I raised regulatory queries (in consultation with the Environment Agency).

108. Hitachi-GE has modelled the swelling of the boron carbide and has developed a failure criterion based on the level of boron depletion. This has been set to a Boron 10 depletion level of [25%], where operational experience indicates that significant release of boron carbide into the coolant would not be expected (Ref. 104). Hitachi-GE advised that the core loading will be such that the boron carbide rods will have enough remaining design life to enable their use for a full cycle of power suppression.

109. Hitachi-GE reports that boron-carbide cladding tube failure does not threaten the geometric integrity of the control rod assembly. I find this argument reasonable.

110. I am aware that the condition of the control rods can be monitored by monitoring the circuit tritium levels and I expect that this will be part of normal monitoring activities, but I do not regard it as an effective primary control measure.

Hafnium Control Rods

111. Structurally, the hafnium rods have a stress limit for irradiation-assisted stress-corrosion cracking, but also the life is limited by irradiation growth.

112. The hafnium rods contain two tube sections. The upper part is [2] mm thick and the lower is [1] mm. These overlap to provide a gap which can accommodate the differential thermal expansion between the hafnium and the stainless steel sheath. This gap is progressively lost due to irradiation. This gap closure is a constraint on irradiation.

113. The limiting stress levels occur at the weld between the rod tip and the tie-rods. A tolerability-of-failure case has been made for this weld (Ref. 105).

114. Overall, I judge that the control rods are designed against a suitable set of design codes and the important degradation mechanisms have been considered by setting limiting values for the design life.

4.5 Neutron Sources

115. A neutron source is needed to provide an adequate count rate for criticality monitoring. This should be ensured, while minimising waste and confining radio-active material.

116. I expect that sufficient measures are taken to ensure confinement of the active material in engineered neutron sources, while ensuring that they provide adequate signals to enable core monitoring.

Requesting Party’s Case

117. Engineered neutron sources are required for source-range neutron monitoring only for the initial core. Hitachi-GE provided an outline of the safety case for the use of neutron sources in Ref. 41.

118. The sources are [californium and palladium]; doubly encapsulated in steel cladding of a suitable material. The active material releases helium by alpha decay and therefore the cladding is subject to some loading due to the build-up of gas. The source is contained in the inner capsule and the outer capsule protects this from corrosion in the coolant environment.

119. The helium gas release in the capsule is calculated to be acceptable over the 60 year life of the reactor, but in practice, the assemblies are discharged after the first cycle because the irradiated fuel will provide a sufficient neutron source for the further operation of the reactor.

120. Eighteen months design life is assumed in order to evaluate the irradiation embrittlement and stress corrosion cracking. The structural integrity is assessed in accordance with ASME design limits and the safety margins are large (Ref. 41).

121. The use of californium provides safety benefits over alternative source designs including very substantially lower radionuclide inventory than alternatives.

ONR Assessment

122. Hitachi-GE use double confinement of the source, which is good practice. In PWR reactors, secondary neutron sources are sometimes required; to ensure a sufficient count rate at the excore detectors. Secondary sources provide an on-going radiological issue, so I welcome their absence in the UK ABWR design (which uses in-core detectors). Since the UK ABWR sources are only used for one cycle of irradiation, I am satisfied that the margin to structural limits resulting from in-service degradation is large.

123. I note that the inventory of radionuclides in the proposed design is orders of magnitude lower than some alternative designs and is therefore a relatively limited hazard (Ref. 41).

4.6 Core Stability

124. SAP ERC.3 advises that core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range.

125. BWRs have a positive pressure coefficient which potentially leads to feedback effects on stability. In addition, there is also the potential for density-wave instability resulting from the feedback between two phase evaporation and frictional pressure gradient. I examined these issues to confirm that the design meets ONR expectations for core stability.

Requesting Party's Case

126. Hitachi-GE's arguments relating to core stability are presented in Ref. 79. Hitachi-GE advises that, BWRs have a large negative power coefficient and this provides inherent stability against disturbances of reactivity caused by various plant operations such as control rod movement. UK ABWR design features which contribute to the stability are:

- the void reactivity coefficient designed not to be excessively negative,
- forced circulation that prevents hydraulic disturbances,
- sintered uranium dioxide pellets with sufficient heat conductivity, and
- introduction of part length fuel pins that contribute to reduced two-phase pressure drop ratio.

127. The UK ABWR has the following improved design features compared to previous BWR plant types:

- A [154.9 mm fuel lattice pitch that is wider than a BWR5 lattice pitch of 152.4mm] (i.e. the UK ABWR has a larger intra-assembly gap). This provides more non-boiling fluid in the core and less negative void coefficient.
- A smaller core inlet orifice diameter compared to the standard BWR5. This results in a larger single-phase to two-phase pressure-drop ratio; which reduces density-wave effects.
- The inner width of the fuel channel is [the same as the early BWR3/4 type fuel and is larger than the latest BWR5 fuel]. This contributes to a lower core pressure drop.
- A larger number of low-ΔP steam separators. This reduces the two-phase pressure drop of the recirculation system.

128. With the characteristics described above, the UK ABWR has sufficient suppression capability against power oscillation within the normal operating domain.

Determination of Operating Space

129. The core stability testing was performed at start-up testing of an existing ABWR for the normal operating region and outside of normal operating region. The results of testing show that the stability performance of UK ABWR is suitable.

130. The ABWR is designed against a criteria defined in terms of the rate of decay of an induced perturbation to the operating conditions. The magnitude of the core and fuel channel decay ratios provide an indication of the propensity of the core power to oscillate in either a core-wide (in-phase) or regional (out-of-phase) mode. The core is required to have a ratio less than 1.0.

131. Frequency-domain analysis has determined the decay ratio for nominal conditions at the most limiting point in the region of permitted operation; in terms of core flow and power. The predicted decay ratio is substantially below the design criteria (Ref. 79).

132. Reactor protection is provided by Reactor Protection System (RPS) scram and selected control rod rapid insertion (SCRRI). This protection acts if the high-power or low-core-flow mode of operation is inadvertently entered.

133. In addition to determining a safe operating regime, fault tolerance has been demonstrated by simulations using the TRACG code.

ONR Assessment

134. I am aware that flow stability has been an operational problem in BWR in the past. This has the potential to lead to fuel damage, but historically, operators have been able to bring these events back under control.

135. I consider the design countermeasures taken by Hitachi-GE in their GE14 fuel design as sensible. I welcome the two-track approach to control:

- Defining operating limits, and
- Setting automatic protection; based on a map of decay ratio targeting a value of [0.8].

136. This is consistent with the principle of defence in depth. I note that the approach involves additional automatic protection for UK ABWR.

137. I chose not to assess the frequency-domain code. This is an old code with an established history of development and application. Hence I considered it more important to focus on the use of TRACG in this context. The use of a system code in this way is relatively novel, so I examined the models and correlations employed. This assessment is detailed in Ref. 47 and is mostly related to fault analysis. However, I looked for specific evidence of the code's capability to represent instabilities observed in BWR plant.

138. A series of experiments have been conducted to examine the code's ability to model density-wave instability. The onset of instability and the decay ratios were predicted with good accuracy (Ref. 80). Hitachi-GE concluded that the results are potentially sensitive to Courant number, but the analysis presented in Ref. 80 do not show evidence of excessive numerical damping, on the contrary, the TRACG results tend to be pessimistic.

139. In conclusion, I judge that Hitachi-GE have taken reasonable measures to mitigate the effects of density-wave instability.

140. I also asked about xenon stability. Hitachi-GE advised that the strong moderator feedback in BWR tends to heavily damp xenon oscillations, so that this is not a significant issue (see for example Ref. 81). I accept this argument.

4.7 Corrosion, Hydriding and Surface Deposits

141. The corrosion of fuel components is of safety significance not so much because it causes loss of metal thickness as because zirconium not only produces a surface oxide film, but it also takes up hydrogen into the cladding metal. This then precipitates as hydrides; which are relatively brittle.

142. The hydride level is significant for fuel ductility; both in fault and operating transients. It can impair the creep strength of cladding during dry storage; and it can affect the dimensional stability of the material.

143. The dimensional stability is important for limiting the fuel assembly bowing and the axial growth of the fuel pins. These effects must be limited so that design constraints are respected.

144. The oxide layer is also important because beyond a certain thickness, it can have an insulating effect; which can lead to cladding failure. I am aware of the fact that early BWR have experienced fuel failures associated with CRUD-induced accelerated corrosion. I therefore asked for information on the control of deposit in UK ABWR.

145. The general control of coolant chemistry is a matter for the Chemistry topic stream, but where these matters relate directly to fuel, we worked together to ensure adequate control and operating limits.

Requesting Party's Case

146. Abnormal corrosion is avoided by tight controls on cladding material composition and the cladding manufacturing process, and through ensuring plant water chemistry falls within industry guidelines (these guidelines are readily achieved in modern BWR plants). It is common practice to perform periodic fuel inspections. However, Hitachi-GE argue that adherence to the measures described above ensures that cladding corrosion performance is within the GNF experience and mitigates the need for routine surveillance (Ref. 15).

147. Heat transfer through soft crud is primarily by wick boiling. Wick boiling is more related to crud structure than crud thickness. The impact of soft crud on fuel pin performance is small but is explicitly included in fuel pin evaluations. Departure from wick boiling occurs when the crud structure is hard and tenacious. The build-up of tenacious crud is dependent on the plant water chemistry.

148. Hitachi-GE argues that the deposition of particulate material on the fuel pins is controlled by controlling the concentration of solutes in the feed water (Ref. 12).

149. The crud-induced cladding failures experienced in the 1970s are thought to be related to copper migrating from the condensers and depositing in the core as a coherent and insulating layer. The use of titanium condensers is expected to reduce soluble copper levels in the UK ABWR to a trace (Ref. 15).

150. Other solutes are controlled by ion exchange in the feed-water system and limits are placed both on key individual solutes and the overall coolant conductivity Ref. 14.

151. Hitachi-GE further argues that GE14 fuel for UK ABWR, with P9 cladding material, has very high corrosion resistance and was developed for resistance to nodular-corrosion related fuel failure. Hitachi-GE reports that this material has never experienced a corrosion-related fuel failure in any BWR, even in those few remaining plants that still have brass condensers.

ONR Assessment

152. The issue of control of crud is a concern not only for fuel integrity, but also for the chemistry topic stream; where the principle concern is the transport of material activated in the core and associated issues relating to radiological releases in normal operation and faults. Hitachi-GE has addressed this issue by selection of primary coolant materials to reduce corrosion products and to reduce the levels of materials with potentially harmful activation products (Ref. 85). This is principally assessed in (Ref. 82).

153. The approach to control of crud adopted by Hitachi-GE is to accept that there will be crud, but to control the solute in the feed water so that the crud does not build up to unacceptable levels and that wick boiling does not come to a halt as a result of solute depositing into the pores of the crud. The approach is partly analytical, but mostly based on experience. I judge this to be a reasonable approach.

154. I discussed operation limits on coolant solute concentrations in joint meetings with the ONR chemistry experts and Hitachi-GE has proposed and justified a set of operating limits. I requested that Hitachi-GE present evidence to support the proposed limits. This has been hampered to some degree by the commercial nature of some of the information used to define the chemistry guidance. However, (Ref. 15) is a significant step in the right direction and I judge that it provides operating rules which will satisfy the requirements of the site license in respect of fuel performance. A wider assessment of these limits is found in the chemistry assessment report (Ref. 82).

155. The data presented on the corrosion of the proposed cladding is for a modest number of fuel pins (Ref. 85) compared to the vast amount of fuel that has been irradiated, but shows generally good performance and, given the extensive operating experience, I am satisfied that it provides adequate justification for loading the fuel. I do not agree with the view that the experience is such that the fuel does not need to be monitored during its irradiation. However, I consider this an issue for a prospective licensee.

4.8 Consideration of Spent Fuel Pond (SFP) Spray System

156. The focus of the fuel and core design is generally on satisfying the requirements of a deterministic safety case. However, I did consider the feasibility of a severe accident mitigation measure; proposed to mitigate the effects of a catastrophic failure for the spent fuel pool .

Requesting Party's Case

157. In the PSA of the spent fuel pool , the reliability of water-level maintenance and cooling systems is evaluated for various postulated faults. If the make-up flow rate is larger than the lost water inventory by leak and/or evaporation, the water level is maintained over the top of active fuel length and fuel failure is prevented. The spray function is not credited in the PSA evaluation because it is conservatively assumed that if the water level cannot be maintained above the fuel, the SFP is damaged.

158. Nevertheless, the objective of SFP spray is to suppress the heat up of fuel and mitigate the excess fuel damage when the water level of SFP decreases abnormally by large leak or other possible causes.

159. Hitachi-GE has evaluated the conditions under which the spray would be effective and reported this in Ref. 18.

ONR Assessment

160. The pool cooling system provides make up water in the event of a leak from the fuel pool; leading to a loss of water level. I accept that anticipated events will probably not lead to a rate of loss which exceeds the capability of these systems, but I welcome nonetheless the addition of spray nozzles capable of cooling the fuel in the event of uncovery. I asked Hitachi-GE to provide information on the capability of these systems and I examined their analysis.

161. The Hitachi-GE model assumes that a quench front descends into the fuel until the spray water evaporates completely or until it reaches the pool water level. The basic assumption is that the rate at which SFP spray water can enter the fuel is governed by a counter-current flooding limitation (Ref. 18). I judge that the physical models applied are appropriate.

162. The model demonstrates that even in the event of a catastrophic failure of the pool, the spray can bring a significant benefit in mitigating radiological releases, and in certain cases, it can prevent fuel damage. I judge that the addition of spray nozzles is a commendable addition to the pool safety systems and is a suitable severe-accident mitigation measure.

4.9 Substantiation of the Validity of Reactor Design Criteria

163. One of the key aspects of the fuel design topic is the definition and quantification of Design Criteria (also referred to as Fuel Safety Criteria). These criteria define the conditions beyond which the functional capability of the particular component can no longer be ensured. These criteria cover normal operation and fault conditions. Examples of are found in Ref. 16, which has informed my views on good practice.

164. My expectations for the protection of the fuel are informed by TAG 075. TAG 075 advises that a set of design limits consistent with the key physical parameters for each structure, system or component is specified for operational states and design basis accidents. The TAG also provides specific examples, which I reviewed for completeness. Certain criteria were assessed in some detail and are discussed below.

Requesting Party's Case

165. The proposed operating rules for fuel are presented in Ref. 43. Design Criteria are presented in Ref. 42. Sets of criteria are identified covering: normal operation; anticipated (frequent) faults; and more extreme Infrequent Fault conditions.

166. The design intent is to maintain the cladding in an operable condition in normal operation and frequent faults and to maintain the functional requirements of the fuel system and the geometry of the core in infrequent faults identified as requiring consideration within the design basis. Hitachi-GE has systematically identified potential failure modes as illustrated in Fig. 1.

167. Technical claims relating to specific criteria are briefly summarised below. Many criteria used are not reported in detail below; on the basis that they are well established and do not merit detailed discussion.

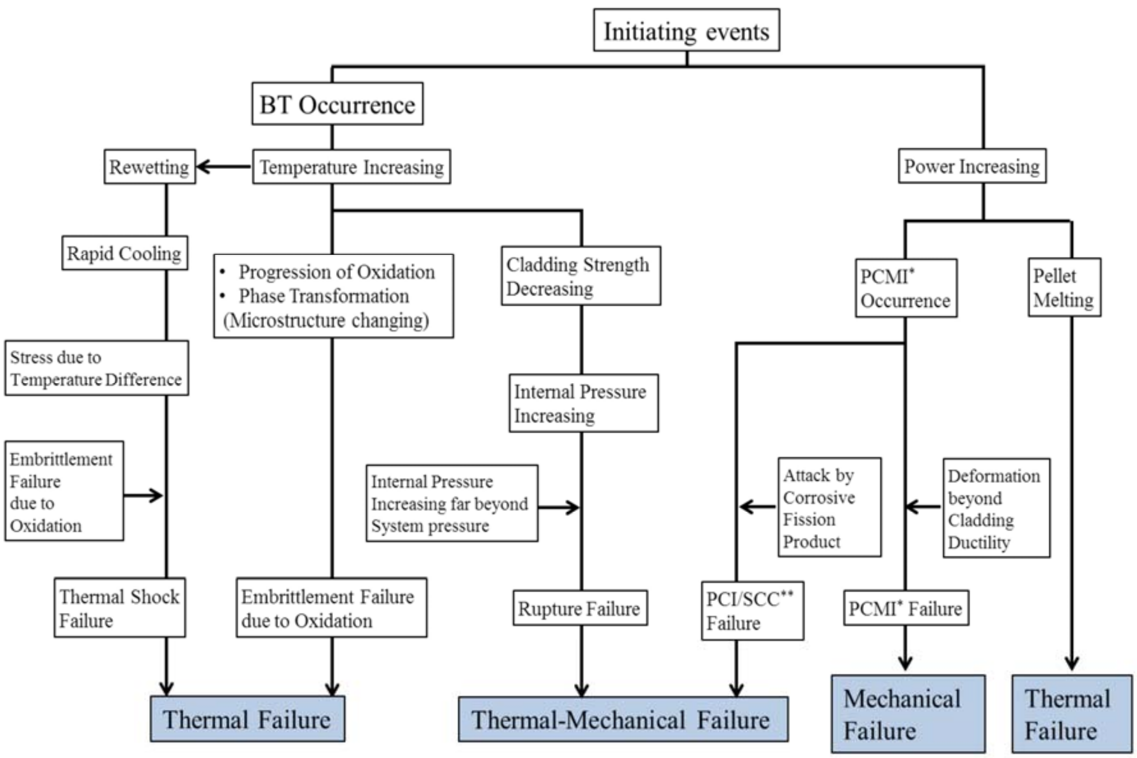

Figure 1, Cladding Failure Modes

BT = Boiling transition

* PCMI = Pellet Clad Mechanical Interaction

** PCI/SCC = Pellet Clad Interaction/Stress Corrosion Cracking

Criteria for Reactivity Faults

168. Hitachi-GE proposes to include criterion for fuel failure and also for loss of coolable geometry and core degradation. These criteria are experimentally based and reflect the main failure mechanisms. Hitachi-GE advised that the safety analysis results for UK ABWR demonstrate that the fuel can be retained below the criteria for cladding failure and therefore the core coolable-geometry degradation criteria are redundant.

RIA Cladding Failure Criteria

169. Hitachi-GE argues that fuel cladding failure mechanisms associated with RIAs include:

- High-temperature failures post-Departure from Nucleate Boiling (DNB) oxygen-induced embrittlement and fragmentation, high temperature cladding creep (rod ballooning and burst).
- Pellet-cladding Mechanical Interaction (PCMI); including hydrogen-enhanced PCMI cladding failure.

170. The PCMI failure threshold is empirical and is justified in Ref. 46. This is based on the increase in Radially-Averaged Peak Fuel Enthalpy (RAPFE).

RIA Coolable Core Geometry Criteria

171. Hitachi-GE recognise various mechanisms which can lead to core degradation and identify four criteria which need to be satisfied to avoid this:

Criterion 1: Peak radial average fuel enthalpy must remain below 230 cal/g.

Criterion 2: Peak fuel temperature must remain below incipient fuel melting conditions.

Criterion 3: Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel pin burst must not threaten the reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

Criterion 4: No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel pin ballooning.

172. However, Hitachi-GE argue that if fuel cladding failure is prevented, then there is no concern with maintaining a coolable core geometry and the criterion related to dispersal of fuel into the coolant becomes irrelevant.

Criteria for core uncovery (or fuel Dryout)

173. For Frequent Faults Hitachi-GE argues that dryout should be prevented, or if not prevented, then cladding integrity should be maintained.

174. For Infrequent Faults, criteria are set down which demonstrate that the cladding will not burst or become excessively embrittled, but failing these, that the structural integrity of the fuel assembly will be maintained; as a means to retain the bulk of the fuel in place.

Criteria for Clad Stress

175. Hitachi-GE has defined cladding failure criteria based on a combination of code limits and ramp-test data. Analysis of frequent fault transients has shown that faults initiated

from steady operation, within the permitted thermal-mechanical limits, are unlikely to fail the fuel.

176. In addition, a combination of analysis and experience has been used to design operating rules for transients that are expected to protect the cladding against failures due to anticipated changes in power shape.

ONR Assessment

177. After several iterations, Ref. 42 provides a systematic and concise justification of the design criteria used in analysis of fuel integrity. I commend the clarity provided in Figure 1.

178. I expect that in accordance with IAEA and WENRA requirements, these criteria should ensure that the fuel integrity is not threatened in normal operation and frequent faults and that, as far as reasonably practical, the cladding retains its confinement function in infrequent accident conditions.

179. My assessment of the completeness of these criteria has been based on consideration of guidance in TAG 75 and relevant practice elsewhere reported by the OECD working group on fuel safety in Ref. 16.

Criteria for Reactivity Faults

180. Hitachi-GE has proposed a set of limits on cladding failure based on Radial-averaged peak fuel enthalpy (RAPFE). The limits are empirical and are based on power pulse tests. The criteria are broadly consistent with practice elsewhere (See for example, Ref. 16) and consider the effect of previous irradiation on fuel performance; both in terms of hydrogen uptake and pin internal pressure. This builds on an extensive programme of experimental work carried out in Japan.

181. The results of UK ABWR fault analysis are below the fuel cladding failure criteria, so fuel-to-coolant interaction is precluded and thus the intent of the structural integrity criterion is satisfied. I am therefore content that Hitachi-GE has made an acceptable safety case based on the available information. However, the international community has judged that more information is necessary and an OECD joint project is underway to obtain this. ONR is a partner in this project.

182. I expect that the results of power pulse tests in the CABRI reactor will be more prototypic than previous tests and I expect that a prospective licensee will review their criteria in line with these results. However, in view of the safety margins demonstrated, I am content for this to be addressed as normal business and I am therefore not raising an assessment finding.

Criteria for core uncovery (or fuel Dryout)

183. Based on TAG 75, I judge that the design intent should be that, where protection is provided against a fault, it should aim to be effective at maintaining the cladding of the fuel intact. Moreover, in more frequent events, the cladding of the fuel should not be damaged; so that the reactor can be returned to power (Ref. 23).

Frequent Faults

184. Hitachi-GE argued for a relaxation to the general practice of preventing boiling transition in favour of limiting the cladding temperature to a brief period at levels below 800ºC. This constraint is intended to prevent any zirconium phase change (which might affect the metallurgy of the cladding) or lead to significant oxidation (Ref. 42).

185. IAEA advises that: to maintain adequate fuel performance margins to dryout of the fuel surface, during normal steady-state operation, additional margin is usually applied to the safety limit which corresponds to the heat flux increase during the worst anticipated operation occurrence (AOO) that might be expected during the plant life. This constitutes an operating limit that is continuously verified during plant operation and analysis of critical power ratio generally used to determine the protection settings (Ref. 5).

186. In the UK, we have similar expectations (Ref. 3), but define Frequent Faults to include events which might be expected during the life of a fleet of such reactors and therefore we include events up to a return frequency of once in a thousand years (1.E-3 /y) of reactor operation. The consequence of this definition is that certain postulated fault sequences associated with secondary system faults are treated as Frequent faults, but Hitachi-GE has not been able to show that the minimum critical power ratio required to avoid dryout will not be exceeded.

187. These faults include power increase associated with the effect of a density wave passing through the core and are too fast to be fully mitigated by reactor protection. Hitachi-GE therefore chooses to argue that the core power density is sufficiently low for the effect on the fuel to be benign. The response to pressure transients is a side effect of the strong moderation density feedback which is generally beneficial in terms of coolant temperature and power feedback and therefore I considered whether the design has been optimised to be ALARP.

188. There is some precedent for accepting the argument that short periods of boiling transition can be tolerated within the UK (in the case of faults relating to loss of electrical supplies). Moreover, I am aware of experimental test performed in the context of loss-of coolant accidents in the UK and elsewhere where fuel experienced quench from elevated temperatures without experiencing loss of ductility (see Ref. 16, 11). Above 800C, there is a significant change to the cladding microstructure, which would make the fuel’s continued operability uncertain, but below this value the issue relates principally to hydrogen distribution and I judge that for a brief (a few seconds) exposure of low-burnup cladding to temperatures up to 800C, a utility would probably ultimately be able to make a case to return the plant (and the fuel) to power operation. I therefore judge that provided a robust demonstration of clad compliance with the proposed temperature limit is made, this is acceptable. I have examined the relevant transients and it appears that the temperature transient is very consistent and typically results in a peak cladding temperature of around 600C (Ref. 119). I judge that this is acceptable.

Infrequent Faults

189. For Infrequent events, I accept that some degradation of the cladding material can be envisaged and that the focus becomes that of maintaining the containment function of the cladding; so that the basic safety objective of limiting release of nuclear material from the fuel pin can be met (in so far as reasonably practical).

190. Hitachi-GE has defined time-at-temperature limits which firstly ensure that the cladding does not burst (releasing fission products and some particulate material) and failing this, limits which ensure that the fuel pin retains sufficient cladding integrity to prevent loss of coolable core geometry and enable fuel recovery. The methodology used to determine dryout conditions is assessed later in this report.

191. In faults leading to fuel dryout while reactor coolant pressure is maintained, Hitachi-GE has defined a limiting cladding temperature as a function of initial fuel pin internal pressure. Criteria have been provided for two periods of exposure to this temperature; 30 minutes and 24 hours. These periods reflect claims made for potential operator actions to mitigate a fault (Ref. 42). The criteria are used to determine the conditions

under which the cladding will fail in intact-circuit faults. They are based on calculations made with the PRIME fuel model. I asked for further information on the models and correlations used to represent high-temperature creep of zirconium alloys. The model is presented in Ref.9 and appears to be satisfactory.

192. In order to provide a criterion for LOCA faults, Hitachi-GE fitted a curve to measured fuel cladding temperature at failure; as a function of cladding hoop stress. They selected data from experiments with ramp rates lower than 5.6 K/s on the basis that this is the adiabatic heat-up rate they expect for BWR fuel uncovered in a LOCA. The collated data covers various materials and ramp rates and is fitted using a least-squares technique (Ref. 42).

193. Hitachi-GE argues that irradiation is observed to have no effect on failure temperature for high-temperature creep (Ref 42). I support this view (although there is perhaps a weak affect around the precipitation of beta phase zirconium) and therefore I judge that the proposed burst criterion is adequate when applied as described.

194. I have considered whether the consequences of ballooning on cladding temperatures have been adequately represented. My concern was partly due to the relatively tight pitch of the fuel pin lattice in the GE14 design.

195. Hitachi-GE argued that any fuel cladding balloons are likely to have a benign effect on fuel cooling. A selection of experimental data was presented including electrically-heated tests on bundles of BWR pins (Ref. 110). I judge that these arguments demonstrated that a coolable geometry can occur, but not necessarily that it will. I therefore asked Hitachi-GE to perform sensitivity studies to examine the effect of a very severe postulated flow blockage on quench (Ref. 111). The results of the simulation demonstrated that once a supply of feedwater is established, the available head of water is such that rapid quench of the assembly is likely to occur. Prior to quench, the heat-up of the assembly was essentially adiabatic. Consequentially, I accept the argument that, for the severe accident sequence analysed, the effect of ballooning is not likely to be significant. No ballooning is calculated in design-basis faults.

Criteria for Clad Stress

196. I considered whether the stress and strain criteria originally proposed complied with the advice in TAG 75; that continued integrity of the cladding be demonstrated for frequent fault sequences. My expectation is that the principal means of protecting the fuel against pellet-cladding interaction (PCI) failure is to design a core such that there is adequate margin (between the steady operating condition and fuel damage) to accommodate faults.

197. Hitachi-GE's constraints on cladding stress were initially based on design-code limits and I was concerned that these did not adequately protect against stress-corrosion cracking by pellet-cladding interaction (PCI). I therefore raised a regulatory query explaining my expectation that protection should be provided based on the performance of the fuel in power ramp tests (as advised by IAEA).

198. Hitachi-GE explained that BWR operators avoid PCI cladding failure by following *Soft Limits Guidance* (Ref. 22) provided by GNF. My enquiries have indicated that this guidance is effectively regarded as mandatory (Ref. 86). I therefore pressed Hitachi-GE to adopt this guidance as operating limits. They have provided Ref. 43 and Ref. 52 which set a basis for operational limits. Ref 43 adopts the soft limits in claim FA SFC 4-10.10 and Ref. 52 requires compliance with the limits in the PCSR. Deconditioning and conditioning rates as well as the maximum allowed step change in nodal power are constrained and when load follow operation with power manoeuvres is planned, Ref.

43 requires the compliance of the soft duty rules with preconditioned envelope should be confirmed prior to the operation.

199. I judge that this is satisfactory. However, the PCSR as a whole is a little confusing in this respect. To ensure that these limits are accorded due importance, I expect the limits to be more explicitly included in higher-level documents in subsequent issues of the PCSR.

200. In response to my expectation, of an adequate demonstration of fault tolerance, Hitachi-GE has defined a limiting fuel linear rating which is to be respected (based on fuel ramp tests) and then used this to define the Thermal-Mechanical Operating Limit (TMOL) function in the core monitoring system. This function limits the steady operating power and Hitachi-GE proposed a limit that leaves a margin of greater than 20% to conditions which caused fuel cladding failure (Ref. 42).

201. Hitachi-GE has demonstrated that a margin of 20% is sufficient to accommodate anticipated frequent faults. They exclude from this consideration of fast reactivity faults; such as those induced by density waves. I judge this reasonable on the basis that this type of transient is too short in duration to permit corrosion-assisted cracking.

202. In the case of planned power changes, Hitachi-GE impose constraints on the rate of power rise to ensure that the thermal stress is relieved prior to a failure threshold being reached, or that the release of corrosive fission products from the fuel is sufficiently slow to prevent reactive chemical species traveling from the release site to the cladding, and causing inner-surface embrittlement (Ref. 87).

203. The fuel local power density has a threshold value (as a function of burnup and residence time in the reactor) below which rates of power change are not constrained. Lower values are applied to fuel that has experienced sustained control-rod insertion or sustained operation at low power. Above these thresholds, power ramp rates (and step changes in power) are constrained. Hitachi-GE argue that for the equilibrium core design, the steady-state power distribution is such that the threshold local power levels at which ratings are constrained are not likely to be reached (Ref. 88).

204. Hitachi-GE report that in BWR, most cladding PCI failures are associated with control-blade swap procedures (used to optimise radial power shape). Power margins to the thresholds might, in principle, be small during control-blade swap procedures, but Hitachi-GE explained that good practice is to carry out these procedures at part power when this is potentially the case.

205. Hitachi-GE argue that in the event of a frequent postulated initiating event (such as loss of feed-water heaters) occurring during rod swap, experience has shown that operators act to reduce power in sufficient time to prevent cladding failures (Ref. 88). This may be true, but since the time taken to achieve a cladding failure by Stress Corrosion Cracking (SCC) is a matter of few minutes, operator action is not to be relied upon.

206. I advised Hitachi-GE that this issue needs to be considered and appropriate time at risk arguments to be made. The two areas potentially of concern are power raise following core reloading and the withdrawal of control blades to expose regions of fuel that have operated for a sustained period at reduced power. Historically fuel failures have occurred during the execution of these manoeuvres and also coincidental operational transients have occurred which could raise clad stress to levels where clad cracking would be a concern. I therefore expect these operations to be carried out under conditions which maintain an adequate margin to cladding failure.

207. It is common practice to perform control rod withdrawal at reduced power when necessary to preserve margins to failure. I expect this to be considered for UK ABWR.

Since this is an operational issue, detailed consideration is outside the scope of GDA. I judge that it is acceptable to address the issue while preparing operating procedures. I am therefore raising an assessment finding to ensure that this is adequately addressed:

> To ensure that the increased risk associated with these operations is adequately managed, the future licensee shall ensure that procedures for carrying out rod swap operation are performed at suitable reactor power levels to maintain adequate safety margin to accommodate frequent fault sequences without damage to the fuel cladding by stress-corrosion cracking.

208. The limits and conditions of operation for cladding power ramp rates are determined partly as a result of fuel-pin thermochemical modelling of the stress-corrosion cracking process. I therefore requested Hitachi-GE to supply documentation detailing the qualification of the modelling of these processes and the uncertainties in the use of this model to predict cladding failure by stress-corrosion cracking.

209. This model has been correlated to Over-ramp and Super-ramp tests and is based on the prediction of clad failure in the experimental results (Ref. 89). I regard this work as a significant step forward in understanding the phenomenon and it increases my confidence in the operating limits proposed.

4.10 Design Criteria for Fuel Storage

210. The wet storage of fuel in pools is generally not a fuel performance issue: The low fuel temperature results in a gap opening between the fuel and the cladding; relieving stress, and corrosion rates are too low to be a concern. There is still a need to maintain chemistry control of the pool water; to limit the potential for stress-corrosion cracking of parts of the fuel assembly and other core components, but this is addressed in other topic areas.

211. Dry storage of spent nuclear fuel is more of an issue in this topic. During storage, fuel cladding degradation is avoided by maintaining adequate cooling in an inert gas environment.

212. The cask outer skin operates in a corrosive environment, so the possibility of it leaking within the storage period is foreseeable and this would lead to loss of the inert gas environment. Should the pins also fail, there is the potential for fuel material to oxidise and powdered fuel to disperse. The design intent of dry storage systems is therefore that the storage system maintains two barriers to the dispersal of nuclear material. These barriers are generally taken as the fuel cladding and the cask. The storage of spent fuel is itself an assessment topic (Ref. 65). However, the assessment in Ref. 65 is dependent on fuel safety limits and these have been considered in this report.

Requesting Party's Case

213. Hitachi-GE's safety case for spent fuel storage is presented in Ref. 62 and a preliminary safety case for export of fuel to interim dry storage is presented in Refs. 63 and 64.

214. The derivation of design limits for spent fuel is reported in Ref. 48. The report identifies a set of operating limits for the fuel; which provide a basis for interim spent fuel system design. These limits include:

- Fuel irradiation;
- Fuel pin internal pressure and cladding stress;
- Cladding oxide thickness and dissolved hydrogen concentration; and

- Decay heat level.

215. Ref. 48 then considers the relevance of potential fuel degradation mechanisms. Hitachi-GE concludes that plastic instability and brittle fracture are not viable failure mechanisms. Thermal creep, as a result of the fuel pin internal pressure, is assessed explicitly against a material ductility criterion.

Cladding Creep

216. The creep models employed depend on the stress level. There are correlations for diffusion creep and dislocation creep. The model uses the higher of the two rates.

217. The creep model accounts for load shedding from the barrier material to the Zircaloy cladding. The barrier is conservatively assumed to make no contribution to determining the stress level.

218. A [2%] strain limit is assumed, but with recovery over time up to a [10%] value. The recovery rate is modelled with an Arrhenius temperature variation. This under-predicts recovery compared measurements (Ref. 48).

Clad Cracking

219. The potential for delayed-hydride cracking (DHC) is considered. DHC is a cracking mechanism whereby hydrogen can diffuse to a stress concentration and precipitate there. Over time, the hydride precipitate grows and can fracture if the stress is high enough. Once initiated, the crack can propagate through multiple hydride accumulation and cracking cycles.

220. Hitachi-GE argue that DHC can only occur under certain circumstances:

- A flaw must be present that attracts hydrogen that then precipitates as a brittle hydride phase.
- The temperature must be less than a temperature threshold. At temperatures above the threshold, crack blunting decreases the stress intensity factor such that crack propagation does not occur and there is a higher solubility of hydrogen, which makes precipitation more difficult.
- The stress intensity factor must be greater than a threshold value for cracking to occur.

221. In addition to hydrogen precipitated at cracks, hydrogen can be precipitated in the bulk of the material when the concentration exceeds the solubility limit. The orientation of the resulting hydride platelets affects the ductility of the material.

222. The maximum hydride level likely for UK ABWR is modest, but the worst case is considered to be fully-dissolved hydride because this maximises the potential for hydrides to precipitate in a radial direction and therefore promote through-thickness cracks.

223. The limiting design temperature determines the maximum temperature at which radial precipitates will form and the stress level under these conditions determines the likelihood of cladding failure. The stress intensity required to fracture the hydride precipitates at the relevant temperature could not be achieved at the limiting rod internal pressure with credible defect sizes.

224. For completeness, the potential for stress-corrosion cracking was considered. The argument is that the stress intensity threshold value for SCC is similar to that used in reactor operating conditions and far exceeds stresses in interim storage.

Accident Conditions

225. In fault transients, the likelihood is that fuel failure mechanism would be plastic instability induced in creep failure. Hitachi-GE judges that SCC and DHC will not be an issue.

Corrosion and Hydriding

226. In the event that the water is not entirely removed from the storage cask, there is the potential for corrosion of the fuel cladding; leading to embrittlement and failure. Hitachi-GE intends to place control limits on levels of moisture; to have tolerable levels of corrosion in the cladding. The analysis assumes modest metal loss which will not invalidate the stress calculations.

ONR Assessment

227. These topics are principally addressed in Ref. 65. However, they fall into the fuel topic area from the perspective of defining design criteria and modelling of the fuel performance.

228. ONR expectations relating to the safety aspects specific to storage of spent nuclear fuel are detailed in Ref. 24. I requested that Hitachi-GE develop a systematic analysis of the degradation mechanisms which could influence fuel performance during dry fuel storage and I expected that they should produce a set of substantiated design limits which the fuel needs to respect. In view of the significant amount of work involved, I raised a regulatory observation to allow the work to have the necessary visibility in the project (Ref. 67).

229. Hitachi-GE's analysis is presented in Ref. 48. This includes the systematic review I expected. I informed my judgement on the adequacy of the information by consideration of material in the open literature and material obtained from numerous by-lateral engagements (see for example Ref. 66 and Ref. 68).

230. I judge that Hitachi-GE has considered the recognised potential mechanisms for the fuel degradation in normal operation and faults. I noted that the fuel model used to evaluate creep strain represents the main deformation mechanisms by empirical models. I judge that the approach to modelling creep failure is reasonable.

231. Hitachi-GE argues that stress-corrosion cracking and hydrogen-assisted crack growth are prevented, for credible defect sizes, by maintaining the stress intensity below that needed to fracture the embrittled material at the crack tip. The critical stress intensity used by Hitachi-GE is supported by evidence which is not inconsistent with values reported in Ref. 66 and significant margins to the proposed limit are demonstrated.

232. I also considered the effect of temperature transients on the hydride precipitates in the cladding. There is evidence that dissolving and reprecipitating the hydride can cause the orientation of the hydride precipitates to change in such a way that the cladding ductility is impaired. Hitachi-GE proposes stress and temperature limits that are consistent with industry practice and should limit the amount of radially-oriented hydride formed. However, Hitachi-GE applies a similar hydride fracture-mechanics argument in this case.

233. Hitachi-GE concedes the possibility of radially-orientated precipitates, but argues that the stress is unlikely to be high enough to fracture the hydride platelets. This is a plausible argument, but Hitachi-GE does not present evidence to validate their model of the material so it is not enough in itself. However, the limits Hitachi-GE propose for normal storage permit existing data on the ductility of the cladding to be used to provide comfort. Hitachi-GE has also demonstrated that the presence of the cladding

liner material affects the radial distribution of hydride: The pure zirconium preferentially absorbs hydrogen and therefore at the interface between the Zircaloy and the barrier material, there is a region depleted in hydrogen. Moreover, at cooling rates (representative of dry fuel storage) the hydride precipitates in this region are likely to be small.

234. I judge that the liner is likely to make UK ABWR fuel cladding substantially tougher than much of the PWR fuel in storage and therefore I am satisfied with the design limits proposed. This topic is a developing area and I expect a potential licensee to continue to monitor the results of current research and to take appropriate action as part of normal business.

235. During bilateral discussions with other regulatory bodies, I became aware of concern relating to pellet swelling; caused by accumulation of helium bubbles in the fuel material. I raised this with Hitachi-GE and they updated Ref. 48 to include analysis of the topic. They confirm that for UK ABWR, the swelling will not be sufficient to close the gap between the pellet and the cladding and therefore this swelling will not impair the containment function of the fuel pin cladding. A generic study by the US NRC reaches a similar conclusion (Ref. 112).

236. Hitachi-GE has presented preliminary analysis of the thermal conditions expected for an example cask design. The results are found in Ref. 69 and Ref. 70. Hitachi-GE advised that they do not wish me to assess the thermal analysis methods used during GDA because the selection of the cask design (and therefore the analysis method) is a matter for the licensee. I therefore simply note that analysis exists to support the use of established dry storage casks.

Accident Conditions

237. Hitachi-GE argued that in the event of a cask failure, by stress-corrosion cracking, the defect in the cask wall would remain small and therefore the inert gas atmosphere would remain for an extended period; allowing time for mitigation measures to be taken. This argument is qualitative and I judge (based on typical daily atmospheric pressure variations experienced in the UK) that even a relatively small hole, would result in loss of the inert gas in a few days. I therefore expected Hitachi-GE to confirm that design limits would not be exceeded in an air atmosphere. Ref. 70 confirms this and therefore I am satisfied that the scope of the calculations is reasonable. However, Hitachi-GE asked that qualification of the analysis method should be out of scope of GDA because the analysis is illustrative only.

238. The thermal design of the fuel cask is an important part of selecting a cask design and this potentially impacts on the civil design in the pool area. It is also not straightforward, so I am raising an assessment finding to ensure that the detailed design of this aspect is given appropriate attention. I propose the following:

> To establish an adequate set of analyses to support the safety case for the established dry storage design, the licensee shall substantiate that the fuel assembly dwell time in the spent fuel pool required prior to transfer to interim storage is consistent with the design requirements of the selected storage cask and associated systems. In particular, the qualification of this analysis will need to demonstrate that the complex physical processes taking place in the cask are adequately represented in the modelling.

Corrosion and Hydriding

239. I support the approach of placing a limit on the available water on the basis that it controls the potential hazard associated with clad metal loss (and associated hydrogen uptake). The criterion is reasonable and the allowable water inventory in this context is quite high. I therefore do not regard this as a major risk.

4.11 Substantiation of Core and Instrumentation and Protection

240. TAG 75 advised that the Inspector should verify that adequate provision has been made for fuel and core monitoring; to ensure that functional requirements are met.

241. In modern reactor designs, ex-core instrumentation is increasingly replaced with in-core instrumentation which gives better spatial resolution of the power distribution. This is sometimes displayed in the control room as detailed power maps. The ability to diagnose faults in the power distribution is significantly enhanced by this practice, but experience has shown that the operator can be misled by faults in such a system. Where these systems are used to support significant safety claims, they need either to meet the requirements of appropriate safety classification or diverse means of monitoring need to be provided, with surveillances at appropriate time intervals.

Requesting Party's Case

242. The neutron monitoring uses both fixed and movable detectors. The fixed in-core detectors are evenly spaced in the axial direction to provide indication of neutron flux levels.

243. The Traversing In-core Probes (TIP) are movable detectors that can traverse vertical core locations to obtain a continuous indication of local flux levels (a TIP trace). TIP assist in fixed detector calibration.

244. In the UK ABWR there are two systems of fixed detectors; covering 9 decades of flux. This domain is divided into a start-up and power range. There is some range overlap. The start-up (intermediate-range) instrumentation has approximately two gain ranges per decade and the range switching takes of the order of [100ms].

245. As well as a reactor period trip, the protection system includes a power-level trip at 15% thermal rated power. This provides protection against reactivity faults at zero power. The power-level trip can be vetoed after the reactor reaches 5% power; consistent with PWR practice. The main difference is that the practice for BWR is not to conduct a low-power flux map using the TIP.

246. The control and instrumentation system includes diverse Class 1 and 2 protection systems, together with a Class 3 Plant Computer System which includes a high-fidelity Core Monitoring System.

247. The protection systems provide mitigation of design-basis faults and the Core Monitoring system provides means for the operator to monitor the safety margins, to comply with limiting conditions of operation, and avoid fault conditions.

ONR Assessment

248. Firstly, I satisfied myself that there is sufficient protection in place to protect the core against reactivity transients and then I considered the adequacy and resilience of systems to provide core monitoring.

Neutron Monitoring System (NMS)

249. The system has to meet different challenges depending on whether an event occurs at shutdown or during power operation. I therefore considered the systems under both these conditions.

250. Hitachi-GE claim that the Control rod worth is limited by Rod worth minimizer (Class 3) to limit rod withdrawal at part power to levels where criticality is not possible (Ref. 114). The withdrawal of control rods is also disabled manually when not in *Startup* or *Operation* mode by the protection-system mode switch and also when the detector signal drops below a specified level (Ref. 33).

251. These measures together are intended to reduce the likelihood of an inadvertent criticality to a low level. However, they are not fully independent of the Class 3 rod control system. Hitachi-GE assign frequencies to the various failure modes of the rod control system; making some fault sequences low frequency. This is not consistent with my expectation for a deterministic safety case for a digital system and therefore irrespective of the frequency arguments presented, I needed to satisfy myself that there is adequate protection against the most severe common-mode failure of this system. This I did in collaboration with the fault study and C&I topic area; through a series of regulatory queries. My expectation is that an adequately robust method of protecting against criticality should be provided; together with a dedicated backup system should this fail.

252. When the flux level is in the source range, the primary means of protection is likely to be monitoring the reactor period (rate of change of flux) (Ref. 32). This should be able to terminate most reactivity transients (rod withdrawals) before the reactor achieves criticality and doses to operators working in the containment would be tolerable (Ref. 39).

253. At the low end of the source range, the signal is subject to a degree of statistical noise and is masked to some extent by background signals therefore trips are not provided on the basis of the rate of rise of neutron signal. Instead, the reactor is tripped on the signal level (set at a level well below that of measurable heat generation). This is switched to a reactor-period trip at an appropriate signal level to allow power ascension. In the interval between the lowest signal (determined by the strength of the neutron source) and the source-range flux level trip, there is the potential for the reactor to become increasingly supercritical. I therefore queried the potential for high levels of net reactivity to be achieved.

254. Hitachi-GE has not formally analysed all-rod withdrawal from this condition for UK ABWR, but advised me that the analysis of the rod withdrawal fault assumed a relatively high initial power level; based on their experience of analysis for UK ABWR in Japan, that higher power levels lead to a worse fault. There are only a few decades between the expected initial signal level and the setpoint for the protection, so this argument is plausible. This is assessed in more detail in the fault analysis topic report (Ref. 114).

255. Should the intermediate range reactor period protection fail during power ascension, once the power range level is reached, effective protection is also provided by the (diverse) power-range monitoring system. Hitachi-GE claims the high-power set point on this system; set at [15%] RTP during start-up mode (Ref. 33).

256. In the case of a fault during power operation, the source-range protection is no longer active and so the protection is principally provided by the Local Power Range Monitoring (LPRM) system sensors. These sensors provide input to the core monitoring system and are also used in diverse processing units to provide average power range monitoring signals (APRM).

257. Protection is provided on rate of power change and on signal level. These signals provide trip and signal levels from adjacent detectors are used to limit control-rod movements.

258. The LPRM signals provide excellent redundant coverage of the core and the APRM provides a degree of diversity. However, in view of the use of common sensors, I considered whether any diverse instrumentation was required.

259. I judge that given the large number of sensors (208) and their use in a sophisticated core monitoring system, the likelihood of a fault in the majority of these sensors going undetected is low. I also consulted with my colleagues in Fault Analysis and they pointed out that over–power faults are also protected by down-comer liquid level instrumentation; which detects sustained mismatches between steam and feed flow rates thereby providing a diverse means of protecting against sustained power increases.

260. On this basis, I judge that it would not be reasonable to require further power-range instrumentation.

261. I asked Hitachi-GE to justify the values selected for the minimum start-up range count rate. This is to ensure the operability of the source range detectors.

262. In response, Hitachi-GE advised that the signal to noise ratio of the detectors in count mode is acceptable at the minimum design count rate. However, the target count rate for design purposes is set higher by a significant factor (Ref. 41).

Axial Power Shape

263. I noted that in contrast to PWR, the protection Hitachi-GE offered did not include a trip on axial power shape. I was concerned because of the very strong effect of moving UK ABWR control rods on local power levels. It also became apparent that a slow uncontrolled insertion of all the control rods could induce changes in power shape sufficient to fail much of the fuel by stress-corrosion cracking. Hitachi-GE advised that there are interlocks within the rod control system to prevent such an action, but ONR took the view that there was the potential for these measures to suffer from a common-cause failure of the control system. I recognised that there was likely to be a need for additional protection and therefore I raised RO-ABWR-0077, requiring Hitachi-GE to demonstrate the adequacy of protection for pellet-cladding interaction in response to control-rod movement faults.

264. Hitachi-GE responded by proposing an addition to the APRM signal processing to introduce a reactor trip on axial flux difference; the Axial−Peaking Power Range Monitor (A-PPRM) (Refs. 116, 117 and 32). I was satisfied with Hitachi-GE's proposal, noting that detailed design work is required and other protection options are not foreclosed at this stage. The assessment of Hitachi-GE's response is found Ref. 118. I am satisfied that the signal from the neutron monitoring system will be adequate for monitoring purposes.

4.12 Core Monitoring System

265. The UK ABWR core monitoring system (CMS) is an impressive combination of reactor physics analysis tools and digital instrumentation. It takes input from an extensive set of in-core fission-rate instruments, which form part of the Local Power-Range Monitoring (LPRM) system. It monitors the constraints placed on the fuel linear ratings and the margin to fuel dryout.

266. The signals are processed and displayed to the operators for control purposes through a complex digital Plant Computer System. I expect that the safety claims made on

such systems are commensurate with their safety class and that reasonable levels of reliability are assumed.

Requesting Party Case

267. The core monitoring system is described in Ref. 36, with detailed formal claims provided in Refs. 34 and 35. The system utilizes the same 3D neutron-diffusion model employed by Hitachi-GE for core design. Inputs from the Process Computer system are used to provide estimates of power distributions and thermal limits and display margins to the following parameters in the plant control room:

- safety limit minimum critical power ratio;
- core power distribution;
- Maximum Linear Heat Generation Rate;
- core flow distribution; and
- the gain constants for LPRM protection calibration.

268. If certain instrumentation data is missing or judged to be in error, calculated values are substituted.

269. The system is safety class 3 and is not the sole system necessary to prevent or mitigate a design basis fault.

ONR Assessment

270. Based on my experience with similar systems, I judged that it would not be practical for the core monitoring system to meet more than the requirements of a Class 3 system. I therefore asked Hitachi-GE to provide a justification of the safety categorisation of the system and of measures needed to demonstrate adequate fault tolerance.

271. Hitachi-GE responded by providing a failure mode analysis (Ref. 37). This identified that failure of the system had the potential to lead to faults which would result in a call on the protection systems.

272. Hitachi-GE argued that postulated software error types would include application software errors and operating system software errors. However all error types lead to monitoring function loss which is bounded by an identified design-basis fault and in each case, for frequent faults, at least two diverse protection systems are available to prevent core damage. Consequentially, the reliability claims placed on the CMS system are limited.

273. One particular area of concern is the calibration of the protection system. It would not normally be acceptable for information from a Class 3 system to be used in a Class 1 system. However, Hitachi-GE identified additional administrative measures which a licensee should consider to mitigate the risk of errors in calculated data used to calibrate the LPRM gain constants. Hitachi-GE advised that to meet UK relevant good practice, the checks by the C&I engineer for LPRM and APRM calibration may be supplemented through independent checks by a suitably qualified and experienced nuclear engineer. This aspect of independent verification will be analysed further in the site specific phase of the project.

274. We observed that the analysis of CMS failure modes did not consider in detail the potential for faults occurring coincidently with unrevealed failure of the core monitoring system (placing the plant outside the normal operating conditions). Hitachi-GE

responded by advising that the risk could be limited by routine surveillance of key parameters using diverse means – to be defined by the licensee.

275. The importance of the CMS system to maintaining the plant within its safe envelope is such that I consider it necessary to raise a finding ensuring that ONR is involved in the development of suitable diverse checking methods. I am aware that diverse means of calculating core power distributions have in the past been instrumental in uncovering errors in core design calculations. International Nuclear Power Operators (INPO) has recommended this as good practice and therefore I am proposing the following assessment finding:

> To reduce the reliance on the Core Monitoring System to levels commensurate with its safety classification, the Licensee shall design and implement an adequate diverse surveillance method to allow the operator to verify that the Core Monitoring System is performing correctly and that the plant is demonstrably within operating limits.

276. On this basis, I am satisfied that the functional requirements placed on the system are appropriate from a core design perspective. Detailed design of the system will be assessed in the control and instrumentation topic area.

Core Misloading

277. I assessed the adequacy of the core monitoring to prevent core misloading. I focused on both the measures to prevent fuel misloading and the monitoring of safety margins during operation of the reactor.

278. My expectations for core misloading faults were based on PWR experience, where the fault is a significant hazard. I requested confirmation that the operator dose in the event of an inadvertent criticality, caused by core misloading, would be low enough to allow continued recovery actions.

279. Hitachi-GE responded by arguing that a misload sufficient to cause criticality is incredible. This is supported by analysis of the cold effective neutron multiplication (K_{eff}) for various misloaded core configurations (Ref. 58). Hitachi-GE demonstrated that a very severe misloading can be tolerated without criticality provided that the control rods remain in the correct place.

280. Hitachi-GE explained that physical removal of the control rod is prevented by geometry until the surrounding fuel assemblies are removed.

281. Hitachi-GE further argued that placing fuel assemblies in a region of the core where control rods are absent is prevented by procedural control and the practice of removing assembly supports at the same time as control rods. Control rods cannot be moved at the same time as fuel because both movements require the fuelling machine (but with different fittings). A control rod exchange programme cannot be completed after fuel movement has recommenced because the presence of fuel prevents the fitting of control rods.

282. The development of adequate procedural controls is principally to be addressed in the Human Factors topic area and is likely to proceed at detailed design stage (after GDA). I judge that in regard of the reactor core, suitable arrangements are likely to be made.

283. There are also hazards associated with misloading fuel into the interim dry storage casks. This is outside the scope of my assessment topic and is addressed in Ref. 60.

4.13 Shutdown systems

284. The SAPs require that at all times there are two diverse means available to shut down the reactor and to control reactivity.

285. The UK ABWR has diverse means of shutting the reactor down with the control rods - both hydraulic and electrical; Fine-Motion Control Rod Drive Mechanism (FMCRD) fast run in. Both means are able to take the core to cold shutdown conditions with a pair of control rods failing to insert. The assessment of the adequacy of analysis used to confirm shutdown margin is reported in Ref. 103.

286. In addition, there is a backup system using soluble boron injection, which has been examined in detail in the fault analysis topic area; to examine the fault tolerance of the reactor to failure of the control rod systems. There is also a Recirculation Flow Runback system which reduces the pump flow rate and therefore recovers safety margin when operating limits are exceeded.

287. Overall, I judge that the requirements of the SAPs are met in this area.

4.14 Management of Failed Fuel

288. The management of damaged fuel in UK ABWR is of particular importance because the release of nuclear material into the reactor coolant has the potential for increasing maintenance doses to operators in the turbine hall and during refuelling outages. It also has implications for release of activity to the environment. In extreme cases, it can lead to difficulties for the plant in conforming to normal nuclear material discharge limits although I have limited my assessment to the nuclear safety aspects. The management of waste is assessed elsewhere. Details of the systems necessary to deliver adequate performance are covered in other work streams; notably Mechanical Engineering, Chemistry and Radwaste.

289. My expectations for the management of damaged fuel are informed by TAGs 075 and 81. TAG 075 advises that the release of activity into the coolant should be detected and procedures followed to ensure that the dispersal of nuclear material is minimised. Furthermore, our initial expectation is that a clean-up system should be employed to control contamination levels unless it can be shown that failed fuel management alone can keep this within acceptable levels. Limits and constraints on coolant activity should be defined as operating rules.

290. A strategy for dealing with fuel failures should be specified. This will generally involve removing the failed fuel so that measures can be taken to mitigate the release of fission products and the further degradation of the fuel structure. The timing of this action will be dependent on compliance with defined limits for activity release and the suitability of measures designed to prevent further degradation of the fuel pin in situ. Good practice is to identify the failed assembly and to retrieve it from the reactor core at the earliest practical opportunity.

291. In some reactor designs, fuel degradation can be minimised by locating the failure and reducing local power levels to relieve the cladding stress. This can be used to delay fuel removal.

292. TAGs 081 advises: The implementation of monitoring and inspection regimes based on identification of reasonably foreseeable failure modes, to detect deviation from expected material condition and develop specific arrangements for dealing with failed or degraded fuel in storage, including the maintenance of any necessary reserved storage capacity for retrieved spent fuel from its store.

293. Consideration is required of the number of barriers to release of nuclear material should fuel or facility fail; and the remedial actions that could be taken to rectify any failure. I expect that in so far as reasonably practical, licensees should provide two barriers to the release of nuclear material where significant dispersal of nuclear material could realistically be expected.

Requesting Party's Case

294. The safety case for the management of damaged fuel is presented in Ref. 13.

295. In technical presentations, the fuel supplier Global Nuclear Fuels (GNF) took the lead. GNF claim that over the last decade it has made significant progress in developing solutions and countermeasures to mitigate or eliminate fuel failure mechanisms. GNF report that it has designed, fabricated, and placed into operation over 169,000 BWR fuel bundles containing over 12 million fuel pins. The experience base in 10x10 lattice designs alone exceeds 51,000 bundles and 4.7million rods and defect rates are extremely low (Ref.30).

296. GNF claim to have identified and characterised each failure mechanism encountered. In each successive design, mitigating actions have been developed and progressively implemented as part of a continuous program aimed toward improved fuel reliability and performance.

297. Today GNF estimate that approximately 90% of all BWR fuel failures in the U.S. can be attributed to a single failure mechanism – debris fretting.

298. Based on the above, Hitachi-GE argues that off-site transport of failed fuel for examination is not common practice for BWR and is out of scope for GDA.

299. GNF argue that fuel damage as a result of operating transients and fault, will be infrequent. However, limited damage to the cladding of the fuel pin; resulting in loss of containment is envisaged within the safety case. Damage resulting in a loss of retrievability, is out of the scope: The handling or repair of such damaged fuel will be evaluated later should it actually occur.

300. GNF have developed detailed guidance for the detection and management of failed fuel in reactor and Hitachi-GE offer a number of options for the storage of failed fuel after discharge from the reactor.

Detection and Management in Reactor

301. Hitachi-GE reports that Boiling Water Reactors in the US, Europe, and Japan utilize the same basic methods to detect the presence of fuel failures (Ref. 13). During normal operation, the failed fuel is detected by the gaseous radiation monitoring system attached to the condenser Off-Gas System. All plants utilize continuous gamma radiation monitoring equipment in the off-gas system piping, augmented by periodic grab sampling of the coolant to measure individual isotopic activities.

302. If a fuel failure is detected, a power suppression test is conducted to locate the failed fuel assembly, and generally the reactor operation is continued with local power suppression of the failed fuel assembly by insertion of one or more control blades in the vicinity of the failed fuel pin (Ref. 13). Hitachi-GE claims that this measure, combined with appropriate procedures to limit power changes, allows full reactor power operation to continue throughout the remainder of the reload cycle with no appreciable increase in off-gas activity (and no secondary degradation of the failed fuel pin).

303. Hitachi-GE claim that it is common for plants to operate 12 to 18 months, or even longer, with a fuel defect and minimal activity release, as well as minimal fuel pellet material release to the coolant.

Failed Fuel Inspection

304. Although power suppression can give reasonable confidence of the location of a fuel failure, Hitachi-GE advises that it is best practice to “sip” the coolant passing through each fuel assembly as it is off-loaded (Ref. 21).

305. Vacuum can sipping offers the highest possible accuracy, but with greater outage schedule impact due to the additional fuel moves (to move bundles into the isolation canisters). Hitachi-GE considers providing this equipment good practice (Ref. 13) although its use will be dependent on the circumstances.

306. After shutdown, a full core sipping test is conducted to identify the failed fuel assembly before or after the transfer from the core to spent fuel pool. Then the failed fuel is placed in the Fuel Preparation Machine, de-channelled, and visually inspected to identify the cause of failure.

307. If the identity of the failed rod cannot be established by visual inspection alone, then it may be necessary to carry out other inspections, such as ultra-sonic testing and fiberscope in order to identify the failed rod.

308. Once the identity of the failed rod within the failed fuel assembly is known, additional inspection could still be needed if visual inspection of the fuel assembly alone did not establish the root cause for the fuel failure. Additional inspection generally would require disassembly of the fuel bundle so that fuel pins could be inspected in detail one rod at a time.

309. If the bundle is of low enough exposure to warrant its reinsertion, the failed rod can be removed and stored in a fuel pin storage container and replaced with another rod (Ref. 13).

Failed Fuel Storage

310. Hitachi-GE argues that failed fuel has been stored in various spent-fuel pools (SFP) without any apparent degradation for over 30 years. This can be explained by noting that the degradation process involves power increases which do not occur in the spent fuel pool. Any release of fission products has been insignificant when the failed fuel is stored in the spent fuel pool.

311. Hitachi-GE argues that Good practice for the storage of failed fuel is that small fuel failures may be stored in the same manner as sound fuel within the SFP, but larger fuel failures may be stored in a container in order to reduce the spread of contamination (Ref. 13).

312. After the required cooling period in the spent-fuel pool, the failed fuel is transferred to the Spent-fuel Interim Store (SFIS). The SFIS storage period is up to 140 years after station decommissioning. The assumption is that the fuel will be stored in a canister similar in design to the Multi-Purpose Canister (MPC) supplied by Holtec. Although detailed design is outside the scope of GDA, Hitachi-GE has identified the need to consider designs of additional containment.

313. Hitachi-GE argues that off-site transportation of failed fuel for detailed post-irradiation examination is not a common BWR practice and is out of the scope for GDA. The main reasons for not undertaking Post-irradiation examination of failed fuel are:

- there has been a vast reduction in fuel failure rates; and

- there have been very few new or unique failure mechanisms observed within the small population of failed rods.

314. GNF has a number of inspections tools for examining failed fuel assemblies. The most commonly used method is visual inspection using an underwater camera with the failed fuel assembly located in the Fuel Prep Machine.

ONR Assessment

315. On the specific requirements of maintaining barriers to release in place, I take TAG 81 as guidance. I interpret it as expecting multiple barriers to the release of activity. Ref. 29 provides further detail on recent industrial practice, which I have used as a source of examples of good practice.

316. My assessment has focused of the adequacy of arrangements made to detect fuel failures and to avoid subsequent degradation of the fuel cladding, which can result in the release of fuel material from the pin. During dry fuel storage, it is my expectation that two barriers to release of activity are maintained as far as reasonably practical.

317. In respect to the pool, I base my judgements partly on my experience with Sizewell B. The fuel used in early reloads experienced a significant number of fretting failures, which led to secondary degradation of the cladding during its continued irradiation. Some fuel assemblies were dismantled and failed fuel removed. After diagnosis of the root cause of cladding failures, a design change was made and the problem has not recurred. Some of the fuel failures have been stored in the fuel pool without introducing excessive levels of pool contamination.

318. I found Ref. 26 a useful review of practice in the industry, although I note that there is no consensus internationally on the requirements for storage of failed fuel in the spent fuel pool. My considerations were influenced by measurements taken in the Paks NPP fuel pool, which indicated that the spread of actinide contamination can be limited and the release of noble gasses from the failure is negligible, compared to other sources.

319. Overall, I accept that the approach adopted by Hitachi-GE of storing fuel with small cladding breaches in the normal way in the pool is reasonable. During wet storage, the mobility of nuclear material is limited by the low fuel temperatures and I acknowledge that experience shows that dispersion of activity can be adequately managed by a combination of a suitable pool decontamination system and by the fuel building active ventilation system.

Detection and Management in Reactor

320. BWR off-gas systems take air in-leakage into the condenser, and non-condensable gases from the reactor (mostly, radiolytic H2 and O2) together with very small traces of fission product gases, to the stack via multiple stages of recombiners, filters, and delay/decay volumes. Several process radiation monitors are present at different points in the system. Xe-133 is, by far, the most important isotope for determining the presence of fuel failures and detection of failures routinely uses the ratio of xenon isotopes present.

321. Hitachi-GE proposed a set of limits on the concentration of activity in the off-gas and I requested more detail, together with justification of the limits proposed. The logic is presented in Ref. 13 and the specific levels are given in Ref. 28.

322. Ref. 13 defines Xe action levels for failed-fuel management. A leak is suspected and monitoring levels increased when background is exceeded by a factor of 5. Action to initiate power suppression testing is taken when power levels exceed 50 times

background and a plant shutdown is initiated when Technical Specification levels are breached.

323. Where fuel cladding breaches are sufficiently large to allow water to enter the pin, there is the potential for steam to attack the cladding material and for hydrogen embrittlement to occur. The early detection of fuel failure and the implementation of measures to limit the stress on the cladding are therefore important for prevention of circuit contamination.

324. There is the potential for loss of control of contamination in such events because circuit contamination can raise background levels of activity and make failure detection less straightforward, so I asked for the introduction of a set of absolute limits on coolant activity to provide confidence that the operator will act appropriately. Hitachi-GE accepted this and their proposals appear logical (Ref. 27).

325. Overall, I am content with Hitachi-GE's strategy for the detection of fuel failure in reactor.

Power Suppression Operation

326. My expectation is that if the plant is to continue operation after a fuel failure becomes apparent, then constraints will be placed on power manoeuvres so as to avoid unnecessary power transients and to limit fuel degradation and activity release. Hitachi-GE has provided detailed proposals to address this issue.

327. One consequence of water ingress into a failed fuel pin is the uptake of hydrogen into the cladding material and its embrittlement. This embrittlement cannot be wholly avoided. However, core power transients can be limited to minimise water ingress and to avoid applying unacceptable stress levels to degraded cladding material. After the identification of an assembly which contains a fuel failure, a control rod is inserted adjacent to the failed fuel and the local power is reduced to relieve the stress on the cladding and therefore to prevent fuel degradation and fission-gas release.

328. Power suppression testing is carried out by reducing the core coolant flow rate (to reduce average power to an appropriate level), then successively inserting control rods to create local power transients (which causes variation in fission-gas pressure in the failed rod and expels some of the gaseous activity). The timing of activity release indicates the location of the failure (Ref. 13).

329. A summary of the guidelines set in (Ref. 13) is given below:

If a failure occurs, perform a power suppression test as soon as practical and locate the defect.
Fully insert a control blade in the leaker cell to suppress the power level of the leaker assembly.
Avoid subsequent leaker power increases:
Re-optimization of the cycle management plan to minimize the leaker's power increases may be required. The desired target is for constant or decreasing power, to the extent practical, at all elevations in the pin, for the rest of the cycle.
Perform control blade movements at reduced power.
Compensate for controlled power perturbations such as turbine control valve testing.
Limit leaker power increases to a slow ramp rate.

330. Hitachi-GE argues that these measures are sufficient to prevent the cladding losing its ability to confine the fuel material as a result of secondary degradation. I welcome this as a significant improvement over PWR practices. However, I asked for an explanation as to why the turbine control valve testing continued to be performed. Hitachi-GE took

the view that since experience has shown no detriment in testing (provided precautions are taken) then the practice should continue. This argument is reasonable, but should be kept under review.

331. I note that Hitachi-GE do not entirely ensure that the cladding is free from stress. Instead they provide guidance on performing power increases as follows:

- Perform necessary leaker power increases by flow control (avoiding changes in power shape).
- Limit the local rate of change of power: Relatively minor power shape alterations such as minor control rod pattern adjustments are necessary, but these evolutions are performed at a low core power (reduced by flow).
- Reduce power, by flow, to offset leaker power increases due to normal testing (e.g. turbine control-valve testing). Return to power, by flow, at an unrestricted (but reasonable) rate.
- Fully withdrawn rods in the extra caution region and all partially inserted control rods in or out of the extra caution region are tested at a power level calculated to insure a margin to the permitted rating envelope; including the effects of the rod insertion.
- The failed fuel envelope is assumed deconditioned* at a rate dependent on local conditions.

332. These precautions seem reasonable and are supported by analysis, but the strongest evidence for their use is a record of success in avoiding degraded fuel failures. I am content with these rules provided that experience continues to prove them effective. I judge that compliance with these rules is necessary to demonstrate ALARP. I therefore asked GNF to detail the rules in their guidance. These are found in Ref. 22. I expect them to be incorporated in station operating rules by a potential licensee. I note that Hitachi-GE have incorporated these rules into Ref. 43 and therefore I am satisfied that this will be achieved without the need of further regulatory intervention.

Failed Fuel Inspection

333. I have considered whether it is reasonable to require arrangements for Post-Irradiation Inspection (PIE) of failed fuel. I accept the view that there is a large body of experience of BWR fuel operation and that for well-established designs, PIE may not be required unless anomalies in the failure rates or characteristics are identified. A similar argument has been made for Sizewell B and examination of failed fuel has been carried out in pool. I therefore accept the argument that routine in-cave PIE of fuel failures may not be required.

334. Hitachi-GE proposes that after each cycle of fuel irradiation, some of the fuel bundles are inspected to confirm their integrity. The fuel channel is detached from the FA for fuel inspection. After inspection, the fuel channel is reattached to the fuel bundle and the fuel is reloaded into the core. They argue that this limited measure is performed to confirm the expected fuel performance is maintained. I judge that this is a reasonable approach. However, this argument restricts the plant to the use of well-established fuel design features and therefore this is an operational decision which the licensee should make. I therefore intend to raise the following finding:

> To ensure that the level of surveillance given to the condition of the discharged fuel is appropriate to the uncertainty associated with the likely performance of

* *As fuel operates at reduced power, the initial pellet-cladding gap is progressively reduced and the fuel reaches a new equilibrium geometry appropriate to the lower power level. This is referred to as deconditioning.*

the refuelling strategy the licensee choses, the licensee shall establish a policy on post-irradiation inspection, including inspection of failed fuel and take account of this in the design of fuel handling equipment.

Failed Fuel Storage

335. In order to ensure that storage and disposal options are not foreclosed during construction of the plant, ONR requested further detail. The strategy that Hitachi-GE has adopted is to keep the damaged fuel within the Spent Fuel Pool (SFP) for as long as possible before it is exported, i.e. to the end of station generation. They argue that this is in line with worldwide practice. During this storage time the SFP systems (cooling and clean up) ensure the condition of the SFP water (with regards to presence of contaminants) is kept within set requirements. Therefore, time is available to operators to determine the most suitable way to export the damaged fuel, and should it be necessary, to design specialised and bespoke equipment to allow for export of the damaged fuel. This is supported by the following arguments (Ref. 25):

- The fuel is much less likely to be at risk of degradation during handling faults after extended wet storage because the decay heat and inventory of fission products will have reduced significantly;
- This policy allows maximum flexibility in engineering suitable equipment for export and permits other equipment to be removed to make space if required;
- Hitachi-GE experience is that the pool clean-up system is able to keep foreseeable levels of operator dose to reasonable levels (this is for the chemistry topic area to confirm); and
- There are a limited number of foreseeable damage mechanisms and the number of damaged assemblies is likely to be low, so that it is not BWR practice to routinely remove failed fuel for post-irradiation examination.

336. The arguments presented are aligned with the practice adopted at Sizewell B and do not seem to be a problem there. I therefore do not think that it would be proportionate to ask for further measures for fuel export.

337. The strategy for SFIS during GDA is to demonstrate that there are a number of potential solutions to handle and store failed fuel of different types within the SFIS system.

338. Hitachi-GE point to a CSNI review of failed fuel practice as an example of good practice (Ref. 26). I agree that this document is useful as a benchmark. It advises that:

The storage of leaking fuel is normally characterised by low activity release (Ref. 38), but in case of large cladding breaches, fragments can fall out from the rods and cause contamination in the storage facility. The transient conditions during storage may result in temporary increase of activity release from the leaking fuel pins.

The vast majority of leaking fuel does not require special handling and is stored in the same manner as intact fuel.

339. Ref.26 further notes that: Many power plants have special containers (canisters, casks) for the storage of leaking fuel pin or fuel assemblies. If the leaking fuel pin can be removed, then the rods are stored in the containers. If it cannot be done, the whole severely damaged assemblies are placed there. This advice is consistent with my experience.

340. Ref. 25 provides a description of each storage option proposed; their advantages and disadvantages, and information to demonstrate that the option could be implemented with the current SFIS approach proposed in the GDA.

341. This approach is a starting point for the development of an ALARP solution to the interim storage of damaged fuel and I accept that the final technology selection is best made closer to the time; when the nature of any issue which may arise with the fuel needing to be stored is better known. I therefore consider this approach acceptable. In order to ensure that this requirement receives adequate attention, I have raised the following assessment finding:

> To provide adequate contingency arrangements for foreseeable fuel damage, the licensee shall review the technology available for the inspection and storage of damaged fuel and provide sufficient equipment and arrangements to ensure that suitable ALARP measures can be taken to meet the requirements for the storage of damaged fuel.

342. Overall, I am content that the measures described to address the issue of damaged fuel are sufficiently mature for generic design acceptance.

4.15 Xenon effects of Load Follow

343. Although BWR plant are capable of fast changes in power level, in practice, power ramp rate restrictions (to prevent fuel PCI failures) may curtail efficient load following operation and lead to significant capacity factor losses during load manoeuvres. Fuel performance considerations are clearly the limiting feature. I therefore chose to examine Hitachi-GE's safety case in this area.

Requesting Party's Case

344. Hitachi-GE's case in relation to load follow operation is described in Ref. 10. Although the BWR is stable with respect to xenon, a significant xenon transient affecting the distribution of core power occurs in each load following cycle.

345. At the lower axial nodes, the xenon-induced axial power shift may lead to problems with the soft duty rules and/or with exceeding linear heat generation rate (LHGR) limits if sufficient margin is not available in the steady state. At nodes above the core mid-plate, xenon axial shift could produce problems with MCPR limits.

346. Experiences with demonstration of a load-follow capability in the USA have had mixed results; with operating limits exceeded and a fuel pin failure.

347. Hitachi-GE has considered various options for UK ABWR compliance with UK Grid Code. The reactor core power will be changed to a limited extent; with further power changes enabled by other functions - such as turbine bypass valves.

348. As a rule, core power reductions below 70% of rated power for load following would require some control rod insertion. Subsequent control rod withdrawal may be limited by fuel preconditioning constraints and therefore this is not the preferred approach.

349. Hitachi-GE proposes to change the recirculation flow according to the grid frequency and change the generator power more quickly by operation of bypass valves.

350. Hitachi-GE computed the core characteristics during anticipated load follow; including a representation of a diurnal load-follow xenon transient. The reactivity was controlled by the core flow only and the control rod position was fixed during the transient.

351. The core thermal margins (MCPR) did not exceed the Operational Limit and the load follow was achieved without introducing a large perturbation of axial power distribution.

ONR Assessment

352. Ref. 10 advises that in base-load operation, the plant has a capability to respond to large grid frequency changes by reducing power, but is not required to increase power. Detailed design changes to the turbine governor control may be needed to meet UK specific requirements, but this will be addressed when detailed design of the controller takes place.

353. Hitachi-GE have considered how the plant could provide a more complete automatic response to grid frequency variations when required and favours providing the response required by the UK distribution grid using of the turbine bypass system. This removes the requirement for the core to undergo significant power transients. There is likely to be a need to make some changes to the bypass system, but this is outside the scope of GDA. I am therefore currently content from a fuel perspective.

354. Hitachi-GE have demonstrated that limited reactor power reductions can be made to respond to diurnal reduction in electricity demand, using variation in the recirculation pump speed and this has a tolerable effect on core xenon distribution (and hence power shape).

355. I support the Hitachi-GE's conclusions from their ALARP study presented in Ref. 10 and I judge that this demonstration of a core capability is sufficient for GDA.

356. I note that Hitachi-GE has little to say on the potential to provide load follow in the event of fuel failures. I accept that this is not likely to arise and I judge that it can be addressed on a case-by-case basis.

4.16 Justification of Analytical Methods, Including Treatment of Uncertainty

357. ONR expects a safety case which demonstrates that the plant operating rules will be adequate to reduce risk to an acceptable level with a high level of confidence.

358. To achieve this it is necessary to demonstrate that the analysis methods are qualified to a level of rigor appropriate to their use. They need to adequately represent the plant and the physical processes important to safety. They need to be tested and qualified against experiment and due account needs to be taken of uncertainty.

359. In order to ensure this, I sampled the qualification of certain computer codes. These were chosen on the basis that they were either used to quantify the more significant risks or where safety margins might not be large compared to the level of uncertainty. The codes selected included: the reactor physics methods used to predict the power shape, the core model in the fault study code used in the more complex fault analysis, the fuel pin model used to predict cladding stress, and the steam explosion model used to qualify the effect of ex-vessel corium-coolant interaction.

UK ABWR Reactor Physics Method TGBLA/PANACEA

360. The purposes of the reactor physics methods are: to confirm that the core design will conform to the operating limits assumed in performance and fault analysis; and to provide data to describe core performance in faults. In addition, the reactor physics codes are used to set the protection limits on minimum critical power ratio by providing pin power factors for use in the thermal-hydraulic assessment. The adequate representation of the core performance is therefore an important part of the reactor design. I examined this in some detail and also commissioned independent confirmatory analysis.

Requesting Party's Case

361. The reactor physics methods used by GE-Hitachi-GE are detailed in Ref. 57.

362. The core performance is modelled in a 3D simulator using simplified macroscopic neutron interaction cross sections; pin power factors and node-edge corrections generated by more detailed calculations. This is in principle a standard approach; necessary to make the modelling of a whole core computationally practical.

363. The generation of the macroscopic reaction-cross-section data is more complex than generally adopted in PWR, in that it utilises a multi-stage approach, where individual pins are modelled in detail employing a 1D transport method and this is coupled to a 2D, multi-group diffusion calculation to determine the power distribution in an assembly.

364. The process involves the following steps:

- A 1D transport calculation for the fuel pin is used to generate thermal cross-sections;
- A homogenised 1D radial transport calculation for the fuel assembly is used to generate epithermal and fast diffusion cross sections;
- A 2D few-group diffusion calculation for the assembly at the level of one node per pin, generates pin powers and macroscopic cross-sections;
- A 3D one-group diffusion calculation models the core as a whole based on the defined assembly loadings, using nuclear data libraries of macroscopic cross sections.

365. Hitachi-GE argue that the method was demonstrated to be correct in terms of its ability to predict reaction rates for fuel assemblies by comparison against more detailed assembly infinite-lattice transport calculations. This argument was supplemented by comparison against measurements for a selection of reactor cores.

Isotopic Pin Depletion

366. The isotopic composition of the fuel pin during depletion is calculated based on a detailed representation of minor actinides and fission products. The temperatures within the pins are calculated based on a detailed representation of the radial power distribution in three groups.

Derivation of Nuclear Data Libraries

367. The fuel assembly depletion calculations are repeated varying a number of boundary conditions (such as channel void fraction) to construct the nuclear data library used in the core simulator. The core simulator interpolates this data to obtain the macroscopic data used to represent each fuel assembly in a specific core loading pattern. This library allows the core model to take account of core power distribution, irradiation history and spectral history.

368. Thermal feedback is provided by a single representative flow channel in each fuel assembly with balancing of pressure losses. This flow is determined by a detailed calculation in each of a number of characteristic channels, from which interpolation is carried out based on power, power shape and crud level.

ONR Assessment

369. I am used to cross-section data for core models generated based on two-dimensional representations of fuel assembly transport calculations, followed by core-wide diffusion calculations in more than one energy group. Where heterogeneous materials are used, in PWR it is often desirable to represent the full core using the transport approximation and to increase the number of energy groups used in the core-wide calculation. I was

therefore surprised that the UK ABWR, with a faster spectrum, was represented using a series of approximations and less spatial detail. I therefore commissioned independent confirmatory analysis using independent methods to confirm that the analysis used by Hitachi-GE adequately represented the physical processes.

370. Detailed calculations for the first cycle (fresh fuel) are reported in Ref. 102. The independent method used the SCALE set of nuclear codes to generate macroscopic reaction cross sections and I judge that the methods used are consistent with general practice for LWR calculations (Hitachi-GE used SCALE for criticality calculations).

371. The SCALE calculations represented the fuel assembly in two dimensions on a pin-by-pin basis and solved the transport equations based on a detailed distribution of isotopes (but with microscopic transport cross sections derived from a representative pin).

372. The depletion of the fuel in a representative assembly was modelled (assuming a uniform power and zero-neutron-current boundary conditions. The results of the GRS and Hitachi-GE calculations were compared and found to agree well; particularly in terms of the peak value of the neutron multiplication factor and the final depletion. This was not what I expected because the actual core depletion shows a systematic trend in neutron multiplication factor over a cycle which the full-core model is not able to represent. The comparison suggests that this is probably not due to the fidelity of the neutron transport model (accepting that both models do not fully represent spectral effects).

373. The first core power distribution was also in good agreement. The error observed did follow a checkerboard-type distribution similar to that of the enrichment distribution. This suggests a systematic difference, but is not thought to exceed the expected level of uncertainty in nuclear data. I asked Hitachi-GE to address this by comparing the measurements of the TIP detectors with the more detailed data from gamma scans of discharged fuel. They demonstrated that these are in reasonable agreement and that the results are consistent with the error budget (Ref. 113).

374. The discrepancies in assembly reactivity were maximum immediately adjacent to the ends of the short fuel pins. In this location there is a very heterogeneous combination of pins and the approximations in the Hitachi-GE method may have been responsible for the difference. I requested that Hitachi-GE advise on the implications of this. They demonstrated that the effect is not significant in terms of channel powers and safety margins (Ref. 107).

375. In view of the complexity of the topic, I asked for a second opinion on the level of uncertainty used in Hitachi-GE calculations. This assessment is reported in Ref. 54 and concludes that analysis of safety margins for shutdown and thermal performance is acceptable. In the documentation provided by Hitachi-GE in support to step 4, the uncertainties concerning shutdown margin and thermal limits calculation are well identified and quantified; consistent with the safety margins claimed.

376. Moreover, a substantiation of the link between the error budgets used in fault studies and substantiation of core models has been provided by Hitachi-GE. Ref. 54 reports that Hitachi-GE demonstrated that uncertainties related to lattice calculation are substantially smaller than the inherent measurement uncertainty of the principal governing core parameters.

377. However, I note that the comparison between the predictions of the Hitachi-GE model and the GRS model are not as good for the equilibrium core as for the fresh core (Ref. 108). It is not entirely clear whether this discrepancy is as a result of discrepancies in the specification of boundary conditions assumed in the comparison or a more fundamental issue; relating to the treatment of spectral history in the nuclear data

libraries. I addressed this issue in a regulatory query relating to spectral effects on isotopic composition. Based on Ref. 109, I am satisfied that the effect is sufficiently small as to be tolerable for the purposes of GDA.

378. Overall, after significant scrutiny of the methods I have concluded that the uncertainties associated with the Hitachi-GE analysis method are indeed lower than I originally expected.

379. I also considered the calculation of criticality in the spent fuel pool. Hitachi-GE used SCALE for this analysis in order to perform pin-wise depletion of the fuel assembly (Ref. 115). The results of the analysis demonstrated acceptable safety margins.

TRACG Transient Core Model

380. The 3D representation of the reactor core has been incorporated into the TRACG code for use in transient analysis (Ref. 78). I have examined the models in this code from a fuel and core perspective. I noted that the one-group representation of the core neutronics would be challenged most by fast, local changes in core reactivity (for example rod-drop faults). This is principally relevant to fault analysis and therefore my assessment can be found in Ref. 47.

381. I judged that from a neutronic perspective, the model in TRACG was fit for purpose.

The PRIME Fuel Pin Thermal-Mechanical Model

382. A thermal-mechanical model is important to the demonstration of adequate performance of the fuel pin in normal operation and fault transients. I therefore expect that the model will be subject to a process of continual development as data becomes available on the performance of the fuel and that it provides a well-qualified model of the fuel performance within its claimed range of validity.

Requesting Party's Case

383. The PRIME03 fuel pin model performs best estimate coupled thermal and mechanical analyses of a fuel pin experiencing a variable operating history (Ref. 56). The model explicitly addresses the effects of:

- Fuel and cladding thermal expansion;
- Fuel and cladding creep and plasticity;
- Cladding irradiation growth;
- Cladding irradiation hardening and thermal annealing of irradiation hardening;
- Fuel irradiation swelling;
- Fuel irradiation-induced densification;
- Fuel cracking and relocation;
- Fuel hot pressing;
- Fission gas generation and exposure-enhanced fission gas release including fission product helium release; and
- Differential axial expansion of the fuel and cladding reflecting axial slip or lockup of the fuel pellets with the cladding.

384. An axisymmetric radial mechanical interaction analysis determines pellet and cladding stresses and strains. The thermal solution is obtained by numerical evaluation of the thermal conductivity integral.

385. Ref. 56 demonstrates the predictive capability of the PRIME model relative to fuel centreline temperature, cladding diametral and axial strains, fuel pin fission gas

release and fuel pin internal pressure. Qualification has been made by comparison of code predictions for relevant parameters to an extensive body of measured data.

Thermal Modelling

386. The temperature distribution in the fuel pin affects material expansion and strain rates and is an important factor in the release of fission products from the fuel.

387. The heat transfer from the surface of the pin takes account of convection and/or boiling processes and the conductivity of heat through prescribed thicknesses of coherent crud and oxide.

388. The heat transfer across the gap between the pellet and the cladding takes account of contact pressure where present, using conventional models and accounts for changes in the gas composition in the pin as fission gas is released, by a weighted average of gas conductivity and surface jump effects. Radiation is accounted for by the use of measured emissivities.

389. The pellet thermal conductivity takes account of the effect of fission products in solution and in gas bubbles. The effect of irradiation damage on fuel thermal conductivity is accounted for based on laser-flash measurements of irradiated fuel material.

390. The porosity of the pellet rim - which has been observed at high fuel irradiation, is accounted for based on measurements of the rim porosity and radial extent.

391. The conductivity model is validated against measurements of centre temperatures made at Halden.

Radial Power Distribution

392. The model used for BWR is the polynomial fit to depletion data. This has been validated against measurements. It has been fitted to a representative depletion at constant void fraction and does not take account of the spectral shift operation, but studies have shown that results are not dependent on this.

Fission Gas Release

393. Fission gas release from the pellet takes account of fuel grain size and allows for grain growth. The model includes diffusive and athermal release mechanisms and represents the temperature threshold for interlinkage of fission gas bubbles on grain boundaries as an irradiation-dependent mechanism for enhanced fission gas release.

Fuel and Cladding Creep

394. Fuel cladding irradiation growth is a fit to data from irradiated light-water reactor (LWR) fuel over the range of interest.

395. Pellet swelling is modelled taking into account fission-product accumulation. The model is based on measurements for modest levels of irradiation and has been examined at higher irradiation.

396. The model assumes plain strain axially except at locations where the tensile stress exceeds the ultimate tensile stress of the fuel pellet, in which case, a crack is assumed to open and plane stress is assumed. Axial slip between the pellet and the cladding is also accounted for.

397. The PRIME code includes a model of pellet fragmentation and relocation. This is semi empirical and is fit to operating data so that predictions of fuel centreline temperatures

and cladding strains are consistent with, and represent best estimate predictions of available experimental data over the ranges of power and exposure.

ONR Assessment

398. In order to ensure that analysis appropriately represents the fuel SAP AV.2 requires that calculation methods used for the analyses should adequately represent the physical processes taking place. This principle is supplemented by SAPs AV.3-8 on the adequacy of models and the use of data. I therefore required sufficient justification of the models employed to facilitate review of their adequacy.

399. Fuel pin models are generally based on a macroscopic representation of the main physical processes; calibrated with the use of data taken from fuel measurements; either in experiments or as part of post-irradiation examination. I have therefore considered the extent to which the qualification of the model is independent from its calibration and the way in which the data has been used to quantify uncertainties and define limits of applicability.

Treatment of Uncertainty in PRIME

400. The treatment of uncertainty is a key part of the safety justification and I found that Ref. 56 includes a clear definition of the conditions for which the PRIME model is qualified and describes the treatment of uncertainty in individual calculations in satisfactory detail.

401. The analysis made to protecting against exceeding the cladding strain limit is performed using a worst-tolerance approach: All design and operating parameters that impact calculated cladding strain are placed at their tolerance limit in the analysis. I consider this to be appropriate based on experience of the success of the approach in preventing fuel failures.

402. More generally, the uncertainty in geometric and operating conditions is accounted for by constructing a linear response surface and determining the most limiting point on that surface at the 95% probability level. This is a standard approach in the industry. Uncertainty in modelling is accounted for by perturbations to key parameters (such as pin power) which have been correlated to the parameter of interest. This is considered in more detail below.

403. Overall, the treatment of uncertainty is considered satisfactory.

Thermal Modelling

404. The surface heat transfer model is a simple one-dimensional convection model in accordance with industry practices. The model then takes account of insulation provided by cladding oxidation and crud assuming a thermal resistance directly related to the oxide thickness. This is a simple model and is consistent with my expectations, assuming that the oxide is of low porosity and that any loose crud does not contribute significantly to the thermal resistance. I judge that this is a reasonable model and this is confirmed by consideration of the ability to predict cladding transient creep, which is temperature sensitive.

405. The modelling of the pellet-cladding gap includes the physical processes that I would expect and is perhaps more elaborate than some in that it accounts for the relocation of pellet fragments using a dynamic model. I judge that the model is plausible and broadly consistent with practices adopted elsewhere.

406. The model of heat conduction within the pellet is based on laser flash test measurements on irradiated fuel and represents the data well up to the claimed

irradiation limit. This model is further qualified by comparison against reactor data from instrumented rods. The pellet rim model appears to represent the observations of high-burnup structure and appears to be reasonable. Overall this model appears to be an adequate representation of the physical processes and this is confirmed by the qualification results, where no significant bias is evident in the data with irradiation or power level. The uncertainty is accommodated by varying the power level until it encapsulates the qualification data. This approach appears reasonable.

Radial Power Distribution

407. I noted that a range of radial power distribution options exist in the code and therefore I issued a query to determine which would be applied for UK ABWR. The approach adopted is to use a function of irradiation derived from reactor physics modelling of BWR fuel pins (Ref. 92, 93). The function appears to be plausible based on my experience of PWR pins of similar radius and the approach is reasonable.

Fission Gas Release

408. The model of fission gas release is important because gas internal pressure is potentially a constraint on irradiation in normal operation and dry fuel storage. It is also used to quantify fission-product release fractions in the event of fuel cladding failure.

409. The fission-gas release model is based on Halden data and widely applied correlations, but differs significantly in the value of the temperature threshold for Interlinkage of grain-boundary voids (compared to the original Halden model). This change is both theoretically justified and consistent with recent data for higher burnup fuel.

410. The model is qualified against pressure in instrumented rods and provides a reasonable representation of gas internal pressure across the range of irradiations anticipated.

Cladding and Fuel Creep

411. The model of the fuel pin solves a finite element representation of the strain field in the cladding and the pellet, with account taken for friction between the pellet and the cladding as a result of axial thermal expansion. The model is slightly more elaborate than normal for such codes and appears to be based on reasonable assumptions. I judge that it is a reasonable attempt to represent the fuel and cladding diametric strain at the pellet centre line.

412. I asked for further details on the representation of cladding stress concentrations in the region of the pellet ends and around open surface cracks in the pellet. The response is provided in Ref. 95. Localized cladding hoop strains and stresses can be produced by pellet-cladding mechanical interaction (PCMI). There are two sources of localization:

- The primary source is radial pellet cracks, which develop due to differential thermal expansion.
- A secondary source of localization is cladding ridging, which occurs at pellet-pellet interfaces because of the tendency of pellets to hourglass.

413. Both of these sources of localization are considered in PRIME, where a stress concentration factor is applied based on more detailed analysis of the effects of cladding-pellet friction and the number of radial cracks in pellet. The value selected is based on pellet geometry before mitigation measures were taken to reduce the effect of hourglassing. This model is employed as part of the cladding fatigue assessment. This approach is essentially the same as used by other licensees and is therefore acceptable in principle.

414. The model of transient pellet swelling includes an empirical factor to account for the accommodation of a proportion of the swelling by the porous rim of the pellet. This accommodation reduces the strain experienced by the cladding and therefore its likelihood to fail. The validity of the model has been qualified by comparison of the predicted cladding strain against the observed stain in power ramp testing. Limiting parameters for the creep model effectively bound the data, which provides satisfactory coverage of the operating regime.

High Burnup Models

415. I have benchmarked the models in PRIME by considering reviews of phenomena in the published literature, for example Ref. 94 and 59.

416. The major high-burnup effects on material properties represented in PRIME are: an exposure dependency in the fuel thermal conductivity (as indicated by results of fuel irradiation programs and laboratory measurements) together with modified irradiation growth for annealed Zircaloy; to reflect observed acceleration in growth at high fluence (burnup).

417. Specific sub-models modified to better predict high exposure experimental data include the fuel pellet relocation and recovery model and the fission gas release model.

418. An increase of pellet porosity is accounted for by a rim model which correlates rim thickness and porosity with local burnup. This results in a decrease of pellet thermal conductivity and an increase of pellet volume. However, the effect of pellet-cladding bonding on heat transfer is neglected and the porosity of the rim is not accounted for in calculating stored energy.

419. I consider that the change in density is negligible because the overall growth of the pellet during its exposure is small. The adequacy of the thermal conductivity model is qualified by comparison with integral measurements of irradiated fuel. The code adequately matches the trend of the data with irradiation.

420. In the event of a fuel pin failure in a high-burnup fuel pin, Hitachi-GE recognises significant uncertainty in the fraction of fission gas released from the rim region of the pellet due to micro cracking. However, I do not consider this phenomenon risk significant because my understanding of the proposed core designs detailed in Refs. 96 and 97 leads me to judge that these pins are unlikely to reach power levels likely to be a concern in this context. I have therefore chosen not to consider this issue in detail.

421. Overall, I judge that the fuel pin model development has made commendable attempts to represent the physical processes occurring as fuel burnup progresses and the PRIME model is appropriate for use in examining fuel safety margins.

Critical Heat Flux Modelling

422. Critical heat flux (CHF) describes a thermal limit where the surface of the fuel pin dries out. Heat transfer across a vapour boundary layer is poor relative to liquid, and exceeding the CHF is often accompanied by a marked increase in the cladding surface temperature. The critical heat flux is a key design parameter for the fuel and therefore a robust treatment of uncertainty is required.

Requesting Party’s Case

423. In the BWR, the margin to CHF is described in terms of the critical power ratio (CPR); the ratio of the (experimental) fuel assembly power at dryout to the actual assembly power. CHF correlations are derived from the analysis of experimental data from

electrically-heated (unirradiated), full-scale fuel bundles tested under laboratory conditions. The correlations make it possible to determine the critical power ratio over a wide range of test conditions, such as pressure and flow rate.

424. Hitachi-GE argues that the CPR limits should be set to ensure that only a very small amount of fuel cladding (0.1% of all fuel pins) is statistically expected to dry out during anticipated operational occurrences. Hitachi-GE argue that this criterion is equivalent to the criterion used in Sizwell B; which demonstrates that the most highly rated pin in the reactor has sufficient margin to the critical power to have a likelihood of dryout of below 5% at the 95% probability level (Ref. 98).

425. Hitachi-GE further claim that it is acceptable for fuel to dry out for a short period of time, provided that the cladding temperature remains sufficiently low to avoid significant oxidation or phase transition of the cladding alloy.

ONR Assessment

426. The widespread use of a 0.1% statistical limit for CPR is confirmed in Ref. 16. However, I do not consider it acceptable to anticipate that it is likely that 0.1% of the fuel may be damaged; in a postulated Frequent event. For Sizewell B, some additional analysis was required to demonstrate that the protection was in fact likely to reduce the expected number of rods drying out to less than one. I therefore examined the level of realism reflected in Hitachi-GE's assessment.

427. The analysis method benefits from simplicity. Experiments are carried out on a full-size fuel assembly and the channel power at which a fuel pin first dries out is measured to a high level of precision. This data is then correlated and the statistical uncertainty of the data is determined (Ref. 99).

428. The measurements do not assume that all the fuel pins in a fuel assembly are at the same power. The sensitivity to radial power distribution is experimentally determined by a perturbation analysis (pin-by-pin) (Ref. 53).

429. I am aware that the sensitivity to fuel assembly radial form factor is potentially dependent on hydraulic parameters (such as coolant mass velocity), so I asked for further information to satisfy myself that the correlation parameters adequately represented the physical processes. I was satisfied that the sensitivity is sufficiently weak not to be a practical concern and that the most safety-significant conditions are likely to be adequately represented (Ref. 100).

430. Hitachi-GE also provided evidence that the power distribution used to derive the protection against dryout is sufficiently conservative that in practice, the likely number of rods drying out is less than one (Ref. 101).

431. On the basis of this information, I consider the method of assessing critical power ratio acceptable.

4.17 Modelling of Molten-fuel Coolant Interaction

432. The interaction of the reactor core melt with water is one of the most complex technical issues in fuel modelling. It involves a number of thermal-hydraulic and chemical phenomena and presents a potentially serious challenge to the reactor vessel and/or containment integrity (Ref. 19).

433. In the event of a severe accident; leading to core melt and reactor pressure vessel failure, the successful containment of the fuel depends on the continued integrity of the containment building. The analysis of specific fault transients is carried out in the Severe Accident topic area to support the level 2 PSA. The fault analysis is assessed

in Ref. 77, but the modelling of the molten fuel and its interaction with water is considered here.

434. I decided not to consider the relevant AUTODYN calculations done by Hitachi-GE because I judge that this aspect is not particularly novel. Instead I focused my assessment on the suitability of the mixing and shock-wave model and the treatment of uncertainty. This is because I had previously observed studies of BWR containment where the uncertainty exceeded the safety margins.

Requesting Party's Case

435. Hitachi-GE provided an outline of the safety case for Fuel Coolant Interaction (FCI) in Ref. 71. A steam explosion occurs when hot liquid contacts with cold water. In this phenomenon, fine fragmentation of the hot liquid causes extremely rapid heat transfer from the hot liquid to cold water. Explosive vaporization, feeds energy into a shock wave and destructive forces can result.

436. The topic of steam explosions is discussed in Ref. 72 and also in response to RQ 272 (Ref. 73). Hitachi-GE argues that in the vessel, the mass of corium that can participate in FCI is limited by the details of the structures in the lower plenum. In addition, the design pressure of the RPV is very large. Therefore, rapid steam generation is not a significant concern for vessel structural integrity. A review of current literature on in-vessel steam explosions indicates that most experts acknowledge the possibility of a damaging in-vessel steam explosion is extremely low. The FCI topic is therefore addressed primarily in the context of ex-vessel explosions.

437. Hitachi-GE recognises that FCI provides two different potential challenges to the containment. The functional requirements are:

- reactor vessel support is not lost due to the impulse load to the pedestal wall; and
- the containment boundary does not fail due to rapid pressurization.

438. The analysis result for rapid pressurization is described in Ref. 72. Rapid pressurization due to FCI can in principle be evaluated by the MAAP code. However, MAAP cannot evaluate whether the supporting function of the reactor vessel is lost due to the impulse load to the pedestal wall by FCI. Therefore, this is evaluated using special purpose methods and tools (JASMINE and AUTODYN).

439. The experimental results obtained using substitute materials do not appear to be prototypic. However, there are sufficient tests with suitable materials. There appears to be significant sensitivity to jet velocity, superheat of the corium and subcooling of the water.

440. The JASMINE model takes corium jet boundary conditions from MAAP. There is some uncertainty in the MAAP prediction of corium superheat on relocation. This affects the corium viscosity and its tendency to break up. However, my assessment focuses on the steam explosion modelling in JASMINE.

441. JASMINE applies three models: the first represents the premixing of the corium and the pool water by simulating the two-phase flow of corium and water, and then the second model represents the shock-wave propagation - based on a finite volume solution; with input from the premixed material state. Finally, pressure-wave interaction with structures is modelled in AUTODYN.

Pre Mixing

442. 1D models are provided for the corium jet and the corium debris bed on the base mat. The jet is assumed to have an undisturbed core of material which shrinks in diameter as the jet accelerates under gravity. The model is qualified against analytical and numerical solutions.

443. Interaction with the water is modelled based on a Lagrangian model of representative particles. Breakup and entrainment of particles occurs based on either the balance between gravitation and surface tension forces, or on a balance between gravitation and inertial forces.

444. The bulk flow of the pool water is modelled assuming two-fluid transport equations for mass, momentum and energy; based on the same interface drag equations as found in TRAC-PF1.

445. Heat transferred from the corium particles is calculated based on an approximate solution to the transient 1D spherical heat conduction equation. The energy is partitioned between the coolant phases based on an input partitioning coefficient.

Shockwave Propagation

446. The energy release in the shock wave is assumed to be determined by a fragmentation process governed by hydraulic shear forces; caused by drag associated with particle acceleration resulting from the action of the shock wave. The fragment rate determines the energy release rate because fine particles reach thermal equilibrium very quickly.

Code Qualification

447. The sonic two-phase flow model was verified by comparison against modelling of simple, one-dimensional, two-phase conditions.

448. The following FCI experiments have been used to qualify JASMINE:

- FARO and KROTOS experiments conducted in JRC ISPRA;
- ALPHA experiment conducted in the Japan Atomic Energy Research Institute.
- COTELS conducted in Nuclear Power Engineering Corporation; and
- TROI conducted in Korea Atomic Energy Research Institute (KAERI).

449. Some of these experiments used a UO2/Zr mixture and some used alumina as simulated corium material. Pressure response (and in some cases, level swell) were compared with JASMINE predictions.

ONR Assessment

450. The models employed are at a complexity level designed to provide physically-reasonable trends and to allow adjustment of the physical parameters to adequately represent experimental data. I examined the basis of the models and found nothing counter-intuitive, except that I noted that the size of the droplets entrained from the corium jet is a user input.

451. The assumption used to partition energy to the coolant phases according to a fixed parameter is fairly arbitrary and will also have a significant influence on the predicted pressure rise. I judge that this model may be reasonable in the early phase of the

transient; where heat transfer is dominated by radiation from the particles to the fluid and the beam length in water may determine the rate of evaporation. The model is likely to predict the long-term void poorly and this is what we see in results presented in (Ref.74). However, Ref.74 demonstrates that the limiting structural loads occur when triggering of a shock wave occurs early in the melt transient, so I do not regard my observation as a serious shortcoming for the present application. Hitachi-GE relies on qualification of this model against integral experiments as detailed above.

452. In the cases presented, the code predicts the pressure rise in a steam explosion generally well; both spatially and temporally. The peak pressure is generally over predicted and the form of the temporal response well predicted (Ref. 74).

453. The code has been compared against a number of experiments of varying degree of representation of a corium release. My assessment of the adequacy of this qualification is based on the principle of a graded approach to safety assessment as advised in IAEA standards (Ref. 75). I regard the likelihood of melt relocation in UK ABWR as low and therefore I judge that the level of qualification provided should be sufficient to give an indication of the level of uncertainty in the prediction of pressures and impulse loads using JASMINE.

454. I take confidence from the participation of the JASMINE users in SERENA (Steam Explosion REsolution for Nuclear Applications); the international OECD programme for the resolution of FCI remaining issues in LWRs (Ref. 75).

455. Broadly speaking, the performance of the JASMINE code was qualitatively similar to that of the majority of the participants and I judge that JASMINE was one of the better codes when compared against measurements. SERENA concluded that:

- Whatever the modelling and numerical approaches, all the participating codes were able to calculate the reactor situations of concern.
- The scatter of the results raises the problem of the quantification of the safety margin for the containment in case of ex-vessel steam explosion.

456. These conclusions are applicable to JASMINE. The material presented indicates that predictions may be slightly conservative, but not sufficient to fully encompass the uncertainty.

457. Overall, I judge that JASMINE provides a reasonable model of the physical processes taking place and therefore meets the requirements of SAP AV.2.

458. In my experience, the uncertainty in modelling steam explosions can have a significant effect on conclusions relating to the success of severe accident mitigation measures. I therefore asked Hitachi-GE to justify its treatment of uncertainty.

459. Experiments indicate that, in the case with high corium jet velocity, there is a tendency to have small particle diameters, high rates of heat transfer to the coolant and large initial pressure rises in the vessel pit (Ref. 71). I therefore asked what particle size Hitachi-GE had assumed in their analysis and how this was justified. Hitachi-GE advised that they selected a particle size as a result of sensitivity studies. The value was chosen to maximise the energy release in the steam explosion (Ref. 76).

460. Hitachi-GE advised that the jet entrainment velocity was selected to best fit the response in representative experiments and the triggering of the steam explosion was assumed to occur at a time when the extent of the melt-water mixture and the water subcooling are maximised - to obtain the largest energy release (Ref. 76).

461. I am satisfied that Hitachi-GE has demonstrated that their model is valid under the conditions relevant to ex-vessel steam explosions in the BWR vessel pit and that due account has been taken of uncertainty in the application of the model.

462. The results to date from these programmes indicate that prototypic core melt shows no potential for a spontaneous explosion. The understanding of the triggering mechanisms is still poor. It has been observed that a steam explosion may spontaneously trigger when the melt comes in contact with structures, and may be artificially triggered by pressure waves induced.

463. To conclude, I judge that the JASMINE code provides a reasonable representation of the physical processes likely to occur in an ex-vessel steam explosion and therefore meets the requirements of ONR SAPs. I judge that the code is suitable for use in assessment of the likely severe accident response, but I caution that there is a high level of uncertainty in the loads on structures. I judge that this is significantly less than an order of magnitude around the predicted value.

4.18 Overseas Regulatory Interface

464. It is ONR's Strategy to engage with key international stakeholders in the UK's existing and future nuclear industry to enable efficient and successful regulation. In accordance with this strategy, ONR collaborates with overseas regulators, both bilaterally and multinationally.

465. In particular, ONR collaborate through the work of the International Atomic Energy Agency and the OECD Nuclear Energy Agency (OECD-NEA). ONR also represent the UK in the Multinational Design Evaluation Programme (MDEP) - a multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities tasked with the review of new reactor power plant designs. This helps to promote consistent nuclear safety assessment standards among different countries.

466. In the Fuel and Core Design assessment, information from bilateral meetings with other regulators has informed my judgement, together with presentations made at the NEA Working Group on Fuel Safety; the NEA Halden programme meetings; and NEA state-of-the-art reports.

4.19 Assessment findings

467. During my assessment 6 residual matters were identified for a future licensee to take forward in their site-specific safety submissions. Details of these are contained in Annex 5.

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

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

- site specific information is required to resolve this matter;
- resolving this matter depends on licensee design choices;
- the matter raised is related to operator specific features / aspects / choices;
- the resolution of this matter requires licensee choices on organisational matters;

- to resolve this matter the plant needs to be at some stage of construction / commissioning.

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

5 CONCLUSIONS

471. This report presents the findings of the Step 4 Fuel and Core assessment of the UK ABWR reactor. To conclude, I am satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for Fuel and Core Design. The fuel design proposed by Hitachi has many years of operating experience. The analysis methods used to substantiate core component design and operating limits is mature. However, significant work has been undertaken during GDA to substantiate to operating criteria.

472. Significant conclusions from the Step 4 Assessment are as follows:

- The GE14 fuel assembly design and the design of related core components are the result of an extended process of design improvements, which reduce the operating risk as far as reasonably practical;
- The Functional requirements of the design have been clearly defined and Design Criteria have been substantiated;
- Compliance with the Design Criteria has been demonstrated using qualified analysis methods and a suitable set of operating rules has been defined.

473. Formal detailed assessment findings can be found Annex 5. My judgement is based upon the following factors:

- Hitachi-GE report incremental improvement in the operational experience with their fuel, resulting in excellent fuel reliability;
- My visit to their manufacturing facility enabled me to witness their high levels of manufacturing quality;
- My sampling of the documentation supporting their analysis methods has satisfied me that the methods employed have a sound technical basis, and Hitachi-GE take appropriate account of uncertainty.

474. I consider that from a Fuel and Core Design view point, the UK ABWR design is suitable for construction in the UK subject to resolution of GDA Issues, future permissions and permits beings secured.

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Annex 1 Relevant Safety Assessment Principles Considered During the Assessment

SAP No	SAP Title	Description
EKP	**Engineering Key Principles**	
EKP.1	Inherent safety	Sets a hierarchy of safety measures favouring passive safety.
EKP.2	Fault tolerance	Requires a benign response to foreseeable events.
EKP.3	Defence in depth	Details the principles of multiple safety barriers.
ERL-	**Reliability Claims**	
ERL.1	Form of claims	Requires an evidence based approach.
ERL.2	Measures to achieve reliability	Required demonstrable safety margins or detection of failures.
EAD	**Ageing and Degradation**	
EAD.1	Safe working life	Adequate margin to the working life should be preserved.
EAD.2	Lifetime margins	Requires inspection and testing to confirm margins.
EMT-	**Maintenance, inspection and testing**	
EMT.1	Identification of requirements	The safety case should define inspection requirements.
AV	**Validity of Data and Methods**	
AV.1	Theoretical models	should adequately represent the facility.
AV.2	Calculation methods	should adequately represent the physical and chemical processes taking place.
AV.4	Use of data	Where uncertainty in the data exists, an appropriate safety margin should be provided.
ERC-	**Reactor Core**	
ERC.1	Design and operation of reactors	Deliver safety functions with an appropriate degree of confidence.
ERC.2	Shutdown systems	At least two diverse systems for shutting down.
ERC.3	Stability in normal operation	The core should be stable when operating parameters go outside their permitted range.
ERC.4	Monitoring of safety-related parameters	The core should be designed so that parameters and conditions important to safety can be monitored and appropriate recovery actions taken.

Annex 2 Relevant Technical Assessment Guides to be Considered During the Fuel Design Step 4 Assessment

TAG Number	TAG Title	Notes
NS-TAST-GD-075.	Safety Aspects Specific to Nuclear Fuel in Power Reactors.	Sets expectations for the safety case in the topic area.
NS-TAST-GD-005.	ONR Guidance on the Demonstration of ALARP,	Informs decision making on compliance with the health and safety at work act.
NS-TAST-GD-042.	Validation of computer codes and calculation methods,	Provides specific requirements for analysis.
NS-TAST-GD-034 Revision 2	Transient Analysis for DBAs in Nuclear Reactors	Explains the required analysis to design the protection against faults and operational transients.

Annex 3 Relevant IAEA Standards to be Considered During the Fuel Design Step 4 Assessment

Reference	Title	Notes
IAEA Safety Standards Series, SF-1, 2006	Fundamental Safety Principles.	Sets Framework.
General Safety Requirements No. GSR Part 4 Rev 1	Safety Assessment for Facilities and Activities.	Sets Expectations for a Safety Case, including a graded approach to safety justification.
Specific Safety Requirements SSR 2/1 2012	Safety of Nuclear Power Plants: Design Safety Standard - Specific Safety Requirements.	Addresses the principle of Defence in Depth, fundamental safety functions and Design-basis analysis as a basis for operating limits.
Safety Guide, NS-G-1.12, 2005	Design of the Reactor Core for Nuclear Power Plant.	Provides a detailed set of guidance useful to ensure completeness.

Annex 4 Relevant WENRA References to be Considered During the Fuel Design Step 4 Assessment

Reference	Title / Description	Notes
WENRA. September 2014.	*WENRA Reactor Reference Safety Levels.*	
WENRA November 2010.	*WENRA Statement on Safety objectives for* new nuclear power plants	
WENRA March 2013.	*Safety of new NPP designs*	

Annex 5 Assessment Findings from the Fuel Design Step 4 Assessment

	Text
AF-ABWR-FD-01	In order to ensure that all reasonably practical measures are taken to reduce risk, the licensee shall at the appropriate time review the operating experience on debris-induced fuel failures to determine whether further mitigation measures are reasonably practical.
AF-ABWR-FD-02	To ensure that the increased risk associated with these operations is adequately managed, the licensee shall ensure that procedures for carrying out rod swap operation are performed at suitable reactor power levels to maintain adequate safety margin to accommodate frequent fault sequences without damage to the fuel cladding by stress-corrosion cracking.
AF-ABWR-FD-03	To establish an adequate set of analyses to support the safety case for the established dry storage design, the licensee shall substantiate that the fuel assembly dwell time in the spent fuel pool (required prior to transfer to interim storage) is consistent with the design requirements of the selected storage cask and associated systems. In particular, the qualification of this analysis will need to demonstrate that the complex physical processes taking place in the cask are adequately represented in the modelling.
AF-ABWR-FD-04	To reduce the reliance on the Core Monitoring System to levels commensurate with its safety classification, the Licensee shall design and implement an adequate diverse surveillance method to allow the operator to verify that the Core Monitoring System is performing correctly and that the plant is demonstrably within operating limits.
AF-ABWR-FD-05	To ensure that the level of surveillance given to the condition of the discharged fuel is appropriate to the uncertainty associated with the likely performance of the refuelling strategy the licensee choses, the licensee shall establish a policy on post-irradiation inspection, including inspection of failed fuel and take account of this in the design of fuel handling equipment.
AF-ABWR-FD-06	To provide adequate contingency arrangements for foreseeable fuel damage, the licensee shall review the technology available for the inspection and storage of damaged fuel and provide sufficient equipment and arrangements to ensure that suitable ALARP measures can be taken to meet the requirements for the storage of damaged fuel.