



**Office for
Nuclear Regulation**

Civil Nuclear Reactor Build - Generic Design Assessment

**Step 2 Assessment of the Fuel and Core Aspects of Hitachi-GE's UK Advanced Boiling
Water Reactor (UK ABWR)**

Assessment Report ONR-CNRP-AR-14-008

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EXECUTIVE SUMMARY

This report presents the results of my assessment of the Fuel and Core Aspects of Hitachi-GE Nuclear Energy Ltd (Hitachi-GE) UK Advanced Boiling Water Reactor (UK ABWR) undertaken as part of Step 2 of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA).

The GDA process calls for a step-wise assessment of the safety submission from Hitachi-GE, the Requesting Party (RP), with the assessments getting increasingly detailed as the project progresses. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including review of key nuclear safety, nuclear security and environmental safety claims with the aim of identifying any fundamental safety or security shortfalls that could prevent the proposed design from being licensed in Great Britain. Therefore during GDA Step 2, my work has focused on the assessment of the key claims in the area of Fuel and Core to judge whether they are complete and reasonable in the light of my current understanding of reactor technology.

For Fuel and Core, my interpretation of the safety claims are:

- The fuel is designed and operated to comply with a set of functional requirements and that safety limits constrain plant operation so that release of radioactive materials remains within acceptable limits.
- High quality and proven design and production processes will reduce the incidence of fuel failures in normal operation.
- The resilience of fuel in faults is assured by analysis of postulated faults against a defined set of fuel design criteria.

I have primarily used ONR's Safety Assessment Principles (SAPs) to judge the adequacy of the claims in the area of Fuel and Core. I have also used ONR's Technical Assessment Guides.

My GDA Step 2 assessment work has involved continuous engagement with the RP in the form of technical exchange workshops and progress meetings. In addition, my understanding of the ABWR technology and therefore, my assessment has significantly benefited from visits to Hitachi Work's Rinkai Works and Omika Works.

My assessment focussed on the RP's Preliminary Safety Report (PSR) and its references relevant to Fuel and Core. The work reported in the PSR and its references can be summarised as follows:

- Control of core reactivity enables the safe shutdown of the reactor under all circumstances.
- Removal of heat produced in the fuel via the coolant fluid can occur in normal operation and faults included within the design basis.
- Containment of radioactive substances inside the fuel clad will occur in frequent faults and fuel will retain a coolable geometry for all reasonably foreseeable faults.

During my GDA Step 2 assessment of the UK ABWR aspects of the safety case related to Fuel and Core, I have identified the following areas of strength:

- The RP has achieved excellent fuel performance in existing ABWR plant during normal operation and I judge that this can be translated to the irradiation conditions proposed for the UK, although this will need to be demonstrated.

- The fuel design limits proposed are in line with my expectations based on my knowledge of international good practice and analysis presented by the RP shows that the design has a good prospect of demonstrating substantial resistance to damage in loss-of-coolant accidents.
- The systematic demonstration of resistance to stresses induced by power changes is a particular regulatory expectation in the UK. In the case of the proposed fuel, this is enhanced by the addition of a soft, pure zirconium liner on the inside of the clad and I am satisfied that satisfactory arguments can be made to support the claim of fuel integrity in frequent faults.
- The modelling of fuel performance is yet to be reported in detail, but based on the material presented to date; I expect it to be in line with my expectations.

Overall, the material presented during Step 2 met the requirements for my Step 2 assessment and forms a suitable basis for proceeding to Step 3 of GDA.

During my GDA Step 2 assessment of the UK ABWR aspects of the safety case related to Fuel and Core, I have identified the following areas that require follow-up:

- A formal justification will be needed to demonstrate that no fuel failures are expected in normal operation and anticipated frequent faults, including adequate allowances for foreseeable fuel degradation mechanisms.
- The distortion of fuel channels has been noted to increase with increasing assembly irradiations, with potentially adverse effects on safety margins. Surveillance and operational constraints are currently used to manage fuel channel distortion. A detailed justification of the adequacy of these measures will be required.
- The fuel irradiation proposed is above that currently practiced in the UK and is achieved by adopting a highly optimised fuel-to-moderator ratio. In particular, the core response to pressure transients can lead to significant short term power transients (including clad dryout). The core kinetic response will require a detailed examination to confirm the adequacy of its safety margins.
- The modelling practices, codes and methods will require additional work including substantiation of safety margins in the context of established levels of uncertainty. To assist my assessment, I have commissioned some independent confirmatory analysis.
- A set of safety limits needs to be developed to ensure acceptable clad integrity in interim storage, taking into account proposed fuel irradiations and all degradation mechanisms.

In relation to my interactions with the RP's Subject Matter Experts (SME) in Fuel and Core, I have found the RP to be competent and helpful. The resources that they are putting into development of the UK ABWR safety case, their responsiveness and openness give me confidence that Step 3 will proceed satisfactorily. Success will be dependent on the availability of detailed technical reports substantiating some of the claims made. Many of these were originally the property of GE and a means of providing this information will need to be established.

I see no reason, on Fuel and Core grounds, why the UK ABWR should not proceed to Step 3 of the GDA process.

LIST OF ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
ASME	American Society of Mechanical Engineers
BAT	Best Available Technique
BMS	Business Management System
BSL	Basic Safety Level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
CHF	Critical Heat Flux
CPR	Critical Power Ratio
DAC	Design Acceptance Confirmation
DNB	Departure from Nucleate Boiling – an event leading to surface dryout
EA	Environment Agency
GEP	Generic Environmental Permit
Hitachi-GE	Hitachi-GE Nuclear Energy Ltd
IAEA	International Atomic Energy Agency
JPO	(Regulators’) Joint Programme Office
LPRM	Local Power Range Neutron Monitor
MCPR	Minimum Critical Power Ratio
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
ONR	Office for Nuclear Regulation
PCSR	Pre-construction Safety Report
PIE	Post-irradiation Examination
PSR	Preliminary Safety Report
RGP	Relevant Good Practice
RHWG	Reactor Harmonization Working Group (of WENRA)

LIST OF ABBREVIATIONS

RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
RRP	Resource Review Panel
SAP(s)	Safety Assessment Principle(s)
SCC	Stress Corrosion Cracking
SFAIRP	So far as is reasonably practicable
SLCS	Standby Liquid Control System
SME	Subject Matter Expert
SRNM	Startup Range Neutron Monitor
TAG	Technical Assessment Guide(s)
TIP	Traversing Incore Probe
TSC	Technical Support Contractor
TSF	Technical Support Framework
UTS	Ultimate Tensile Stress
WENRA	Western European Nuclear Regulators' Association

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Table 3: Items to follow up in Step 3

1. INTRODUCTION

1.1 Background

1. The Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA) process calls for a step-wise assessment of the Requesting Party's (RP) safety submission with the assessments getting increasingly detailed as the project progresses. Hitachi-GE Nuclear Energy Ltd's (Hitachi-GE) is the RP for the GDA of the UK Advanced Boiling Water Reactor (UK ABWR).
2. During Step 1 of GDA, which is the preparatory part of the design assessment process, the RP established its project management and technical teams and made arrangements for the GDA of its ABWR design. Also, during Step 1 the RP prepared submissions to be evaluated by ONR and the Environment Agency (EA) during Step 2.
3. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including review of key nuclear safety, nuclear security and environmental safety claims with the aim of identifying any fundamental safety or security shortfalls that could prevent the proposed design from being licensed in Great Britain.
4. This report presents the results of my assessment of the Fuel and Core aspects of the RP's UK ABWR as presented in the UK ABWR Preliminary Safety Report (PSR) (Ref. 7) and (Ref. 8).

1.2 Methodology

5. My assessment has been undertaken in accordance with the requirements of the Office for Nuclear Regulation (ONR) Business Management System (BMS) procedure PI/FWD (Ref. 1). The ONR Safety Assessment Principles (SAPs) (Ref. 2), together with supporting Technical Assessment Guides (TAG) (Ref. 3) have been used as the basis for this assessment.
6. My assessment has followed my GDA Step 2 Assessment Plan for Fuel and Core (Ref 6) prepared in December 2013 and shared with the RP to maximise openness and transparency.

2. ASSESSMENT STRATEGY

7. This section presents my strategy for the GDA Step 2 assessment of the Fuel and Core of the UK ABWR (Ref 6). It also includes the scope of the assessment and the standards and criteria that I have applied.

2.1 Scope of the Step 2 Fuel and Core Assessment

8. The objective of my GDA Step 2 Fuel and Core assessment for the UK ABWR was to review and judge whether the claims made by the RP related to Fuel and Core, that underpin the safety and environmental aspects of the UK ABWR are complete and reasonable in the light of our current understanding of reactor technology.
9. For Fuel and Core, the "safety claims" are interpreted as being:
 - The fuel is designed and operated to comply with a set of functional requirements and that safety limits constrain plant performance so that release of radioactive materials remains within acceptable limits.
 - High quality and proven design and production processes will reduce the incidence of fuel failures in normal operation.
 - The resilience of fuel in faults is assured by analysis of postulated faults against a defined set of fuel design criteria.

10. During GDA Step 2 I have also evaluated whether the safety claims related to Fuel and Core are supported by a body of technical documentation sufficient to allow me to proceed with GDA work beyond Step 2. While I have not received completed documents to date, I am satisfied that information will be supplied in sufficient time.
11. I have identified topics that require further assessment in Step 3 of GDA and these will be included in the project plan.
12. Finally, during Step 2, I have prepared for my Step 3 assessment by securing a contract with Gesellschaft für Anlagen und Reaktorsicherheit (GRS) in Germany to carry out confirmatory analysis of the core power distribution and kinetic properties.

2.2 Standards and Criteria

13. The goal of the GDA Step 2 assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety, security and environmental case. For this purpose, within ONR, assessment is undertaken in line with the requirements of the Business Management System (BMS) document PI/FWD (Ref. 1). Appendix 1 of Ref. 1 sets down the process of assessment within ONR; Appendix 2 explains the process associated with sampling of safety case documentation.
14. In addition, the Safety Assessment Principles (SAPs) (Ref. 2) constitute the regulatory principles against which duty holders' safety cases are judged, and, therefore, they are the basis for ONR's nuclear safety assessment and have been used for GDA Step 2 assessment of the UK ABWR. The SAPs 2006 Edition (Revision 1 January 2008) was benchmarked against the International Atomic Energy Agency (IAEA) standards (as they existed in 2004). They are currently being reviewed.
15. Furthermore, ONR is a member of the Western European Nuclear Regulators' Association (WENRA). WENRA has developed Reference Levels, which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors.
16. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and enlarged on in Technical Assessment Guides (Ref. 3). These guides provide the principal means for assessing the Fuel and Core aspects in practice.
17. The key Safety Assessment Principles (SAP)s applied within the assessment are SAPs EKP, ERL EAD FA, and ERC (Ref. 2) (see also Table 1 for further details).
18. The following Technical Assessment Guides have been used as part of this assessment (Ref. 3):
 - NS-TAST-GD-005 Guidance on ALARP (As Low As Reasonably Practicable).
 - NS-TAST-GD-016 Integrity of Metal Components and Structures.
 - NS-TAST-GD-037 Heat Transport Systems.
 - NS-TAST-GD-042 Validation of computer codes and calculation methods.
 - NS-TAST-GD-081, Aspects Specific to Storage of Spent Nuclear Fuel.

2.2.1 National and International Standards and Guidance

19. The following national and international standards and guidance have also been used as part of this assessment:
20. IAEA standards (Ref. 4):
 - Fundamental Safety Principles. IAEA Safety Standards Series No. SF-1 , IAEA Vienna 2006.

- Design of the Reactor Core for Nuclear Power Plant, IAEA Safety Guide, NS-G-1.12, IAEA Vienna 2005.
 - Safety of Nuclear Power Plants: Design, Specific Safety Requirements, IAEA Standards Series No. NS-R-1c IAEA, Vienna (2012).
21. WENRA references (Ref. 5):
- Reactor Safety Reference Levels (January 2008).
 - Safety Objectives for New Power Reactors (December 2009) and Statement on Safety Objectives for New Nuclear Power Plants (November 2010).
 - Waste and Spent Fuel Storage Safety Reference Levels (February 2011).
 - Statement on Safety Objectives for New Nuclear Power Plants (March 2013) and Safety of New NPP Designs (March 2013).

22. Key reference levels used in the assessment are detailed in Table 2.

2.3 Use of Technical Support Contractors

23. During Step 2 I have engaged a Technical Support Contractor (TSC) to carry out independent analysis of the proposed reactor core in order to confirm prediction of power distribution and kinetic parameters, although this work will not be completed during Step 2. This independent work will include an assessment of the uncertainty in the core power distribution based on uncertainty in nuclear data and estimates of uncertainty in hydraulic conditions.

2.4 Integration with Other Assessment Topics

24. I recognise that during GDA, there will be a need to consult with other assessors (including Environment Agency's assessors) as part of the Fuel and Core assessment process. Similarly, other assessors will seek input from my assessment of the Fuel and Core for the ABWR. I consider these interactions are important for the success of the project and I have made every effort to identify as many potential interactions as possible between the Fuel and Core and other technical areas, with the understanding that this position may evolve throughout the ABWR GDA.

25. Also, it should be noted that the interactions between the Fuel and Core and some technical areas will need to be formalised since aspects of the assessment in those areas constitute formal inputs to the Fuel and Core assessment, and vice versa. Based on previous GDA experience, these are:

- Radiological Protection: provides input to the criticality aspects of the Fuel and Core assessment. This work will be led by the Fuel and Core Inspector.
 - Reactor Chemistry provides input to the crud corrosion and sensitisation aspects of the Fuel and Core assessment. This work will be led by me in coordination with the Reactor Chemistry Inspector.
 - The Fuel and Core assessment provides input to the Fuel Failure Criteria and core dynamic response aspects of the Fault assessment. This work will be led by Fuel and Core.
26. In addition to the above, there will be interactions between Fuel and Core and the rest of the technical areas, Human Factors, Structural Integrity etc. These interactions are expected to happen continuously during GDA, they will be two-way and, mostly, of an informal nature.

2.5 Out of Scope Items

27. The detailed design of interim fuel storage facilities has been left outside the scope of my GDA Step 2 assessment of the UK ABWR Fuel and Core. The reason for leaving this matter out of the scope of my GDA Step 2 assessment is that the facility will not be needed until approximately ten years after the station starts operation. However, in order to avoid foreclosing options, consideration has been given to the compatibility of the fuel design with potential operating limits. See Section 4.6.
28. It should be noted that the above omission does not invalidate the conclusions from my GDA Step 2 assessment.

3. REQUESTING PARTY'S SAFETY CASE

29. This section presents a summary of the RP's preliminary safety case in the area of Fuel and Core. It also identifies the documents submitted by the RP which have formed the basis of my assessment of the UK ABWR Fuel and Core during GDA Step 2. My assessment of this material follows in Section 4.

3.1 Summary of the RP's Preliminary Safety Case in the Area of Fuel and Core

30. The aspects covered by the UK ABWR preliminary safety case in the area of Fuel and Core can be broadly grouped under five headings which can be summarised as follows:

- Fuel System Design
- Core Nuclear Design
- Thermal hydraulic Design
- Spent Fuel Storage
- Analysis Methods

31. Further details on these topics are given below:

3.2 Fuel System Design

32. The fuel system (comprised of the fuel bundle, fuel channel, channel fastener and control rod) is designed to ensure that fuel damage, should it occur, would not result in the release of radioactive materials in excess of limits prescribed. The PSR concerns itself principally with the fuel bundle.

3.2.1 Fuel Assembly

33. The fuel assembly consists of a bundle of fuel pins supported between upper and lower tie plates. The assembly is surrounded and constrained by a channel box. The upper tie plate includes a handle which is used for lifting the assembly.

34. The entire fuel bundle is held together by 8 threaded tie rods located around the periphery of the bundle. Another key component of the bundle is the set of 8 spacer assemblies that maintain proper spacing of fuel pins along the axial length of the bundle as well as increasing turbulence, which raises the Critical Heat Flux and therefore avoids dryout of the pin surface.

35. Adequate free volume is provided within each fuel pin in the form of a pellet-to-clad gap and a plenum region at the top of each fuel pin to accommodate thermal and irradiation expansion of the UO₂ and to reduce the internal pressure resulting from gaseous fission products liberated over the life of the fuel.

36. The fuel to be loaded in the UK ABWR is designated as GE14. It has been deployed in reload quantities for over fifteen years and has an excellent record of reliability. GE14 has also been selected as the initial core fuel for two ABWRs in Taiwan.

37. The lower tie plate of the GE14 bundle houses a debris filter. This filter provides resistance to debris fretting, thereby substantially improving fuel reliability.

38. The fuel is designed to meet the following system functional requirements, making a distinction between frequent faults (which have a postulated return frequency greater than once in a thousand years and therefore might reasonably be expected to occur world wide within the life of the plant) and infrequent faults which are postulated as reasonably foreseeable, but are less likely to occur:

39. In normal operation and all design basis faults, the following functional requirements apply:
- Control of core reactivity and safe core shutdown under all circumstances.
 - Residual heat removal through preservation of a coolable geometry.
40. In addition, in normal operation and frequent design basis faults, radioactive materials (in particular fission products) are expected to remain contained within the fuel clad boundary.

3.2.2 Control Rod

41. The control rod consists of a sheathed cruciform array of either stainless steel tubes filled with boron carbide powder or hafnium metal.
42. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of control rods to counterbalance steam void effects at the top of the core.
43. The control rod is designed to have:
- Sufficient mechanical strength to prevent dispersal of its reactivity control material.
 - Sufficient strength to prevent deformation that could inhibit its motion.
44. High shut-down reliability is the result of a number of features of the control rod System. For example:
- The accumulator for each hydraulic actuator provides sufficient stored energy to insert two control rods at any reactor pressure.
 - Each pair of drive mechanisms has its own redundant system of valves to increase failure tolerance.
 - The fine-motion control rod drive hollow piston and guide tube are designed so that they will not prevent control rod insertion by other means.
 - Each fine-motion control rod drive mechanism initiates electric motor-driven insertion of its control rod simultaneous with the initiation of hydraulic scram. This provides a diverse means to assure control rod insertion.
45. Two types of rod are used in the UK-ABWR; hafnium and boron carbide. The hafnium rods retain their absorption capability over a number of nuclear reactions and therefore have a longer neutronic life than boron carbide equivalents. Hafnium does not have as large a neutron cross section as boron. Consequentially, the hafnium rods are used for core power shape control and the boron rods for shutdown.
46. The hafnium tube thickness is varied axially as part of measures to reduce the assembly weight and preserve insertion times.

3.3 Core Nuclear Design

47. The safety functional requirements met by the neutronic core design are:
- Control of core reactivity to enable the reactor to be safely shutdown under all circumstances.
 - Removal of heat produced in the fuel via the coolant fluid.
 - Containment of radioactive substances (actinides and fission products).
48. The reactor core design limits constrain the response of the core in power transients and hence help to limit the core to tolerable conditions for the fuel.

49. The maximum reactivity worth and the reactivity insertion rate of the control rods are small enough to ensure that anticipated reactivity insertion events do not damage the reactor coolant pressure boundary, and do not cause damage to the core, core support structure and pressure vessel internal structure that may impair cooling of the fuel.
50. The ABWR control rod system is designed to provide control of the maximum excess reactivity anticipated during plant operation. The shutdown capability is conservatively evaluated assuming a cold, xenon-free core.
51. The reactor shutdown system is able to bring the core to a subcritical state from any power operation condition. It has two independent systems: The control rod and fine-motion control rod drive system; and the Standby Liquid Control System (SLCS).
52. The control rods (and fine-motion control rod drive system) are able to bring the core to a subcritical state at hot or cold conditions when the pair of control rods with the largest reactivity worth (specifically, one rod or a pair of rods belonging to the same hydraulic control unit) are completely withdrawn out of the core and cannot be inserted.
53. The Standby Liquid Control System can shutdown the core and maintain a subcritical state at a hot stand-by condition.
54. In addition, the core response itself tends to limit power changes:
55. As moderator density decreases, the neutron leakage from the core is increased (affecting the chain reaction) and the control rod worth becomes greater (tending to shut the reactor down). These two effects always have a negative feedback on the core power.
56. The moderator temperature coefficient varies depending on temperature and core burn-up, but the effect on reactivity is small and not significant.
57. In normal operation, the core pressure is maintained at a constant value. As a result, the coolant temperature is constant regardless of the power level, except for minor variation in the sub-cooled zone. The void fraction varies depending on the power level and the core flow rate.
58. The RP claims that power oscillations are readily detected and suppressed, should they occur.

3.3.1.1 Design Criteria

59. To satisfy the design bases, the following specific items for the design are considered:
 - The overall moderator void coefficient is negative.
 - The Doppler coefficient is negative and has sufficient reactivity feedback characteristics for acceptable performance in infrequent design basis faults.

3.3.1.2 Misloading

60. An identifying fuel assembly serial number is engraved on the top of the handle; no two assemblies bear the same serial number.

3.3.2 Core Xenon Stability

61. The RP claims that special power oscillations caused by local xenon fission-product transients have not been observed in operating BWR and both testing and analysis have demonstrated that xenon oscillations in BWR are heavily damped due to strong negative power feedback on reactivity.

3.4 Thermal hydraulic Design

62. The permissible design limits of the fuel shall be established with the aim of preventing fuel damage. The following shall be considered as mechanisms of fuel damage:
- Perforation of clad by overheating caused by insufficient cooling.
 - Perforation of clad by strain caused by relative expansion of clad and fuel pellets.
63. Overheating is prevented by ensuring that the Minimum Critical Power Ratio is respected. The critical power ratio (CPR) is defined as: the ratio of the critical power (bundle power at which some point within the assembly experiences the onset of boiling transition) to the operating bundle power. The intention being to maintain a margin between the predicted conditions and the condition under which the fuel surface would dry out, so as to accommodate random variation and uncertainties.
64. Conditions in frequent design basis faults (caused by a single operator error or equipment malfunction) shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, fuel clad integrity would be assured.
65. The minimum allowable critical power ratio is set to correspond to the criterion that the vast majority (99.9%) of the pins are expected to avoid boiling transition.
66. Fuel channel bow influences the thickness of the water gap between bundles and therefore impacts the pin power distributions inside the bundle. The pin power distribution influences the bundle R-factor, thereby impacting the bundle CPR. The R-factor is a parameter that characterizes the radial power peaking in the bundle. Therefore, the R-factor used in cycle specific core design and core monitoring calculations is dependent on the amount of channel bow. The amount of channel bow applicable to a specific core design is determined using a channel bow prediction model.
67. The ABWR has the following design features which promote hydraulic stability compared to the previous BWR plant types:
- A fuel lattice pitch that is wider than a BWR5 lattice. This provides more non-boiling area in the core and less negative void coefficient.
 - A smaller core inlet orifice diameter compared to the standard BWR5. This makes the flow less sensitive to two-phase pressure drop.
 - The inner width of the channel box 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-resistance moisture separators reduces the two phase pressure drop of the recirculation system.

3.4.1 Neutron Source Assemblies

68. The current submission does not provide information on the design limits for neutron source assemblies.

3.5 Analysis Methods

3.5.1 Assembly Structural Modelling

69. The initial submission is silent on assembly structural design, but in response to my query RQ-ABWR-0041, the RP advises that the structural design is based on finite-element models, implemented in a commercial package, against the requirements of

the American Nuclear Society mechanical engineering standard Ref. 16 and US federal regulation 10CFR 50 (Ref.18).

3.5.2 Fuel pin Modelling

70. The fuel pin thermal-mechanical evaluations are all performed using the PRIME03 fuel pin thermal-mechanical performance model.
71. The PRIME03 fuel pin performance model performs best estimate coupled thermal and mechanical analyses of a fuel pin operating history. The model explicitly addresses the effects of identified physical processes affecting the fuel.

3.5.3 Core Neutronic Modelling

72. The analysis of the core response is carried out using a sequential process:
73. Three-dimensional representations of a fuel assembly are used to model neutron transport on the basis of a calculated neutron energy spectrum, yielding macroscopic reaction cross-section data for the assembly.
74. The core reaction rates (and hence power) are calculated using the macroscopic data.
75. The core response model is used to define parametric data for simplified core models used for analysis of postulated faults and potential core transients.

3.6 Basis of Assessment: RP's Documentation

76. The RP's documentation that has formed the basis for my GDA Step 2 assessment of the safety claims related to the Fuel and Core for the UK ABWR is:
 - UK ABWR Preliminary Safety Report PSR Chapter on Fuel and Core (Ref. 7). This document describes the fuel, its functional requirements and the basic claims made to demonstrate that these requirements can be substantiated.
 - UK ABWR Initial Safety Case Report on Spent Fuel Pool (Ref 8). Sets down the claims for spent fuel storage.
 - Responses to Regulatory Queries.
 - RQ-ABWR-0024, Fuel Channel Bow Allowances.
 - RQ-ABWR-0025, Fuel Spacer Growth Allowances.
 - RQ-ABWR-0038, Core Loading Design.
 - RQ-ABWR-0039, Repair of Failed Fuel.
 - RQ-ABWR-0040, Control Rod Design Limits.
 - RQ-ABWR-0041, Specific Fuel Design Codes and Standards.
 - RQ-ABWR-0042 Critical Power Analysis Criteria.
 - RQ-ABWR-0043. Core Design Data for Confirmatory Analysis.
77. Details can be found in TRIM folder 5.1.3.9389.
78. I have also recently received UK ABWR pre-construction safety report (PCSR) Chapter on Fuel and Core (Ref. 19). This is based on the PSR, but contains additional useful material and a substantial body of references to other material. I will assess it during Step 3. Copies of the main topical reports are expected in August 2014 and I have seen some of the content informally.

4. ONR ASSESSMENT

79. My assessment has been carried out in accordance with ONR How2 BMS document PI/FWD, "Purpose and Scope of Permissioning" (Ref. 1).
80. My Step 2 assessment work has involved continuous engagement with the RP's Fuel and Core Subject Matter Experts (SME); ie, Technical Exchange Workshops (2 in Japan and 1 the UK) and progress meetings (mostly video conferences) have been held. Related information is found in TRIM folder 4.4.1.2145.
81. I have visited:
- Kashiwazaki Kariwa Units 6&7 ABWRs where I toured the majority of the facility including the upper drywell and the fuel storage pond.
 - Omika Works and Hitachi Works where I saw manufacture of the control rods, channel boxes and the reactor internals.
82. During my GDA Step 2 assessment, I have sought clarification of technical detail by the issue of regulatory queries (RQs), I have raised approximately 9 RQs. No shortfalls in the safety case have led to the issue of regulatory observations (ROs). However, I am considering raising an RO during GDA Step 3 on design criteria for dry fuel storage.
83. I present the details of my GDA Step 2 assessment of the UK ABWR preliminary safety case in the area of Fuel and Core including the areas of strength that I have identified, as well as the items that require follow-up and the conclusions reached in the following sub-sections. I have examined the following against my expectations:
- The fuel and core will continue to function as required during all anticipated operating states and modes.
 - There is a definition of the operating envelope and suitable safety margins exist.
 - The fuel and core will have acceptable performance in faults.
 - The development of suitable operating rules and practices to support the above claims.
 - There is a basis for the substantiation of codes and analysis methods employed to support the above.

4.1 Fuel System Design

84. The SAPs require that the design and operation of the reactor should ensure that the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor. In particular, there should be suitable and sufficient margins between the normal operational values of safety-related parameters and the values at which the physical barriers to release of fission products are challenged.
85. The RP proposes to operate the fuel up to a high level of discharge irradiation and therefore I have focused on the coverage of potential degradation and failure mechanisms. These have been described in Ref. 13 as follows:
- Excessive Corrosion and hydriding;
 - Manufacturing defects;
 - Pellet Clad Interaction (iodine assisted stress corrosion cracking);
 - Clad collapse;
 - Fretting; and
 - Excessive fuel assembly bowing.

86. These issues are addressed below when the performance of the system components are considered.

4.1.1 Corrosion

87. As burn-up increases, corrosion rates are expected to increase as a result of changes to the microstructure of the clad and the uptake of hydrogen. The RP claims that analysis, experimental evidence and Post-irradiation Examination (PIE) data, show that fuel pin failure from hydride embrittlement cracking can be prevented.
88. The RP has made presentations on this topic, including joint presentations with experts from the chemistry topical area and have shown that the proposed Zircaloy 2 clad has a low corrosion rate, with levels of dissolved hydrogen at discharge well below levels that require control.
89. The RP provided information that, in 10 by 10 fuel assemblies similar to GE14, there have been no corrosion failures in reactor. However in step 3, I will examine conditions under which this exposure took place to determine the applicability of this experience to the expected conditions for the UK ABWR.
90. I expect that the RP will propose control limits for corrosion and hydriding and an appropriate programme of surveillance to support the fuel performance claims made. This is particularly the case if interim dry storage of the fuel is likely to be required. See Section 4.4.
91. I note that corrosion can be enhanced by excessive crud formation on the surface of the clad and I am aware that the RP proposes to take measures to limit crud formation by limiting primary-circuit corrosion rates (Ref. 21). These seem reasonable in principle.
92. In collaboration with my chemistry colleagues, I will need to examine this topic in more detail in Step 3, particularly the extent to which proposed operating conditions remain within current experience (including local heat flux levels).

4.1.2 Manufacturing Defects

93. A systematic approach to elimination of defects has led to a very low rate of failures in service. This topic is not addressed in the PSR submission. I asked technical query RQ-ABWR-0039 on the proposed approach to managing fuel failures.
94. The main cause of failure is currently debris fretting, often caused by poor practice in excluding foreign material from the core. The RP has developed a lower tie-plate design to trap the majority of this debris before it reaches the fuel and they propose to use this design within the UK. This is in line with my expectations.
95. During our technical meetings, the RP presented an approach to mitigating the effects of a fuel pin failure without immediately discharging the fuel. A damaged fuel pin within an assembly is found by inducing local power transients and monitoring the effect on primary-circuit activity. The local power level is reduced by inserting a control rod. This limits clad stress and prevents a clad defect from opening further.
96. I am aware that some BWR operators pursue a policy of early removal in a forced outage, but if the defect is very small and growth can be limited, it may be possible to make ALARP arguments otherwise. I expect the RP to present these arguments in more detail in Step 3.

4.1.3 Pellet-Clad Interaction (PCI)

97. The RP claims that under abnormal conditions (including the maximum overpower condition) the maximum fuel-pin heat generation rate will not cause fuel melting or cause the stress and strain limits to be exceeded. This claim supports the more general claim on the preservation of fuel integrity in normal operation and frequent faults.
98. There is currently insufficient detail in the submission on this topic, but the RP has provided me with further information in presentations (Ref. 12). IAEA standards documents recommend that clad integrity in power transients be demonstrated by measures including power ramp tests on fuel pins. The RP presented the results of extensive power ramp-testing and demonstrated that there is a substantial margin between the limiting stress levels expected in normal operation and those at which clad failure may be anticipated. They propose to demonstrate that this margin is sufficient to accommodate all faults anticipated as frequent. This is strong evidence of the robustness of the design.
99. The RP further advises that it has mitigated PCI by using soft zirconium metal as a liner on the inside of the clad. Typically, the liner can reduce the peak hoop stress by 20%, but the chemical inertness of the barrier is the strongest effect.
100. The manufacturing criterion for missing pellet surface (including handling damage) is set such that the stress concentration is limited to a level similar to that of a normal irradiation-induced crack in the pellet.
101. Flow control and fine-motion Control Rod Drive Mechanisms are also helpful in controlling power shape.
102. This fuel protection strategy has been supported by operating rules to limit the rate of power change; allowing stresses to relax and aggressive chemical species to dissipate before damage is accrued. PCI failures have not occurred for 10 years.
103. I have no concerns with the approach described to me provided that it is suitably documented and justified. I will assess the written arguments in more detail when presented during Step 3.

4.1.4 Fuel Pin Fretting

104. On the basis the claims made on operational experience and testing in Ref. 19 and my knowledge, I accept that current BWR design and operation has reduced this problem to an acceptable level. I will not focus on this topic in Step 3.

4.1.5 Channel Box Design

105. The design limits for the fuel channel box are not addressed in the PSR submission. However, I requested information in RQ-ABWR-0024. The channel box is a zirconium sheet fabrication, stiffened at the corners by varying the gauge of the sheet. It is optimised to meet the competing demands of neutronic and structural performance, and suffers some distortion as a result of irradiation creep during its design life.
106. The box can bulge as a result of hydraulic pressure differentials and bow because of differential rates of irradiation creep and hydrogen pickup.
107. Distortion of BWR fuel assemblies is a fleet issue; impacting power distribution and control-rod insertion (Ref. 14 and 15). The move to Zircaloy 2 in the USA reduced corrosion generally, but caused shadow corrosion problems on the surface of the boxes adjacent to control rods; causing hydrogen-induced growth and bowing of the

box at high burnup. In Japan, utilities continue to use Zircaloy 4 at limited discharge irradiations.

108. For ABWR, the initial clearance to the control rod is well within the operating experience. The recent increase in this gap improves nuclear performance (stability) and mechanical margins in this respect.
109. Control rods withdraw checks are performed on blade movement every 7 days. Scram time testing is carried out on individual rods (at part power). The practice of control blade insertion swaps and the placement of fresh fuel have also reduced the problem.
110. I judge that an acceptable solution to this problem can be identified for UK-ABWR and I will examine the proposed strategy in more detail in Step 3. I expect ALARP arguments supporting the strategy to be documented, and suitable evidence to be provided.

4.1.6 Tie-plate Design

111. The upper and lower end assemblies of a BWR channel are referred to as tie plates. They have an important role in ensuring the structural integrity of the fuel assembly and in directing the coolant flow.
112. The PSR did not cover the structural design of the fuel tie plate, so I asked for further information. Structural analysis is carried out against criteria in American Nuclear Society mechanical design code (Ref. 16) (which are similar to those in ASME standards). Finite-element analysis is carried out using an ANSYS™ beam model. ANSYS™ is a widely accepted industry-standard analysis tool and therefore will not be the focus of my assessment.
113. Loads on the handle, exceed the material yield stress but there is margin to the ultimate tensile stress (UTS) (Ref. 12). This is acceptable in design standards for off-normal conditions. The approach appears reasonable in concept.
114. The upper tie plate and fuel transfer handle are a single stainless steel casting. The handle and eight tie rods support the load of the assembly. Design loading is based on a multiple of the fuel assembly weight as a handling limit. Tie rods use three times the fuel weight. These values are consistent with industry practice for fuel transport acceleration limits. However, in the case of the top nozzle, loads caused by the fuel handling machine need to be considered and the hazard of a dropped fuel assembly avoided.
115. There is monitoring of the fuel handling machine load and it is fitted with a trip; designed to act in the event of a snag. I will examine the adequacy of this allowance in Step 3, including data on operational experience and analysis of the inertia of the fuel handling machine in the event of a snag.
116. The lower tie plate is fabricated from a number of castings, assembled to accommodate the filter matrix. Impact loads consider a moderate acceleration consistent with fuel transport limits used elsewhere. This is again appears reasonable and I will not sample this aspect in Step 3.
117. The fuel-pin support grid is modelled by Finite-element analysis. The calculation shows significant margin to yield stress. I am satisfied that it has been subject to seismic load testing to beyond UK requirements.

4.1.7 Control-rod Design

118. The nuclear life of a control-rod is based on worth reduction. After the minimum worth is reached, the control rod is replaced. However, for the boron carbide rods, the

mechanical life is more limiting. This is limited by the internal pressure of the clad tubes and swelling of the boron carbide powder. Near the end of the design life, the clad tubes start to fail and boron carbide washes out. This causes some release of tritium into the coolant.

119. I will examine the substantiation of the control rod irradiation limits and the ALARP arguments that support them.
120. Irradiation-assisted stress-corrosion cracking (SCC) has been observed in control rod sheaths, but not in regions of safety significance. It will be necessary to justify the resilience of the design to this degradation mechanism in critical areas of the assembly.
121. These aspects will be assessed in Step 3.

4.2 Core Nuclear Design

122. The SAPs require that the core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their specified range. Furthermore, a suitable and sufficient shutdown margin should be maintained at all times. This includes the requirement for diverse means of shutdown and adequate shutdown margin to hold down reactivity post trip in the event of degradation of components of the shutdown system.
123. In the UK-ABWR design, the control rods can be inserted both by hydraulic and electrical means and the rod worth is sufficient to take the core to cold shutdown conditions with a pair of control rods failing to insert (for example after the failure of a control unit).
124. The worth of control rods inserted into the core during power operation is automatically limited to ensure that the consequences of a rod drop or withdrawal of a rod bank are acceptable.
125. These provisions address the high-level requirements for shutdown. I will examine the analysis methods used to set the design parameters for the rod worth limiting control system during Step 3.
126. The design of the fuel assembly has been modified to increase the fraction of water that bypasses the fuel so that the moderator void coefficient is reduced and the core reactivity is less sensitive to pressure fluctuations. However, this is balanced by proposed fuel enrichment at the upper limit of what can be manufactured without special measures to avoid inadvertent criticality. This balance has resulted in a large void coefficient and a substantial reactivity response to reactor pressure changes. The RP argues that this is acceptable for anticipated operational occurrences and faults.
127. In Step 3, the RP will present analysis to demonstrate that the reactivity response to anticipated pressure transients is acceptable. In consultation with my colleagues in the fault-study topic area, I will examine the core modelling used to demonstrate this to satisfy myself that this has been demonstrated to a high level of confidence. I intend that the confirmatory analysis I have commissioned will inform this assessment.

4.2.1 Core Monitoring

128. The neutron monitoring system is composed entirely of well-distributed in-core fission chambers through the core (Ref. 12). The system feeds scram and control-rod freeze logic.
129. Fixed in-core fission chambers (LPRMs) provide continuous local power range neutron flux monitoring. A guide tube in each in-core assembly provides access for a Traversing In-core Probe (TIP) providing high-resolution axial detail. Start-up Range

Neutron Monitors (SRNMs) are located at fixed locations between the LPRMs. In source-range mode, the system monitors count rate, and in intermediate and power range, the signal is proportional. The instrumentation has automatic gain adjustment in the intermediate flux region, with continuous signal during transition.

130. The coverage and functionality of this system appears to compare well to some PWR, particularly given the relatively large neutron diffusion length in BWR during power operation.
131. All instruments are fission chambers and are replaced after depletion. The automatic traverse flux map (TIP) system is used for calibration of the fission chambers. It is accommodated in the same instrumentation tubes.
132. Life of fission chambers appears to be based on the depletion of the chambers. I will write an RQ requesting justification of this criterion.
133. In respect to the potential for detection of fuel misloading, I judge that misloads could potentially take the form of either batch enrichment errors or assembly placement errors.
134. The fuel assembly must be oriented properly in each fuel cell to ensure proper fuel bundle power distribution is achieved as a result of enrichment loadings. Improper fuel assembly orientation results in asymmetric power production which may affect the operating margin to thermal limits. I have examined this issue and I am satisfied that, any error would be readily detected visually and therefore this is not a significant risk.
135. In regard to binary-swap misplacements, the RP has advised that the worst case misloading can be detected by coolant activity measurement after fuel has failed. This is not in line with my expectations because I believe that further measures are reasonably practical. For Sizewell B, the in-core flux measurement system has been shown to detect all binary-swap misloadings that perturb the core power distribution sufficiently to impede heat removal within the bounds of operation permitted by the protection system. ABWR would appear to have similar instrumentation. I expect a demonstration that the core monitoring is ALARP when compared against this example of relevant good practice. I will follow this up as part of Step 3 assessment. In view of the available instrumentation, I judge that a satisfactory case can be made, but it will require detailed analysis to define suitable operating limits and rules.

4.2.2 Neutron Source Assemblies

136. Neutron assemblies are not covered in the preliminary safety case. The necessary safety justification will be required in Step 3. I have not examined the design limits for these components during Step 2. I will ask a Regulatory query during Step 3.

4.2.3 Core Xenon Stability

137. I accept the RP's claim that the power coefficient for ABWR is likely to be sufficiently large to limit xenon oscillations. I note that the pre-construction safety report references published material to support this claim. Were this not to be the case, it would become evident to plant operators.
138. I am satisfied that core xenon stability is not a safety issue I need to address for ABWR and that I do not intend to target this issue during Step 3.

4.3 Thermal hydraulic Design

139. SAP EDR.4 requires that no single random failure, assumed to occur anywhere within the systems provided to secure core cooling, should prevent the performance of that

safety function. This includes fractures of cooling pipework and failures of control systems.

140. The RP claims that the UK-ABWR has sufficient safety injection systems to be resilient to design-basis loss-of-coolant faults, without drying out the fuel. This topic is therefore primarily the concern of the fault studies topic area and will not be addressed here. However, faults relating to reductions in flow and increases in power remain a topic for assessment because they require consideration of the detailed core performance.
141. The ABWR thermal power and core flow conditions have certain restrictions because of overall plant control characteristics, core thermal power limits, etc. This power-flow map illustrates the power range of operation used in the system response analyses. The nuclear system equipment, nuclear instrumentation, and the Reactor Protection System, in conjunction with operating procedures, ensures that the core operates within the area of the map defined for normal operating conditions.
142. The objective for normal operation and frequent design basis faults is to maintain fuel clad integrity. The design criteria utilized to demonstrate satisfactory cooling is the critical power ratio (CPR). CPR is the ratio of the critical power for fuel pin dryout to the operating bundle power.
143. The generation of the Minimum Critical Power Ratio (MCPR) limit requires a statistical analysis of the core near the limiting MCPR condition. The MCPR Fuel Clad Integrity Safety Limit applies for not only core-wide frequent design basis faults, but is also applied to the analysis of asymmetric faults such as rod withdrawal.
144. The minimum allowable critical power ratio is set to correspond to the criterion that 99.9% of the pins are expected to avoid boiling transition. This differs from the conventional approach in PWR, which is to demonstrate that the most limiting pin in the reactor has a power level such that it has a 95% probability of not exceeding the critical heat flux at the 95% confidence level. The RP advises that these criteria are broadly equivalent. In Sizewell B, there is a requirement that (in addition to most limiting pin criterion) a demonstration is made that the first line protection would be able to prevent the expected number of rods exceeding the critical heat flux reaching one. I judge that these criteria will in practice be broadly similar, but have not yet examined this topic in sufficient detail.
145. My expectation is that the RP will be able to demonstrate with a high level of confidence that no fuel damage will occur for all frequent faults including those fuel pins which experience conditions systematically different from the bulk of the population of pins. In particular, I expect robust arguments to be made to demonstrate that uncertainties associated with pin-power variation in an assembly are adequately represented. I will examine the arguments made by the RP on this topic in Step 3.
146. I understand from our discussions that the RP proposes to relax the conventional fuel design criteria for certain frequent faults from a critical heat flux (CHF) to a fuel temperature criterion. The motivation for this is that the trip of all Reactor Internal Pumps is postulated as a frequent fault due to the potential for common-mode failure of the control system and initial indications suggest that the CHF will be exceeded.
147. The RP argues that the heat flux will be sufficiently low that no fuel damage is expected. They anticipate that the event will cause the peak clad temperature to reach approximately 600C. This is not expected to cause degradation of the fuel or clad material. There is some precedent for the use of this argument, for example, in German PWR where some fuel is predicted to dry out briefly during the pump coast down and at Sizewell B in the event of an excess steam demand with consequential loss of 11kV supplies. The RP proposes a peak clad temperature limit of 800C to retain alpha-phase crystal structure of the clad and hence to enable mechanical handling and

storage. Assuming that the temperature limit is respected, the temperature transient would anneal the clad and therefore should the event occur, I do not anticipate fuel failure provided that clad creep rates can be kept acceptably low.

148. Subsequent to the fault, the allowable fission gas pressure would be reduced due to the lack of irradiation hardening and therefore a fuel operability assessment would be required before continued power operation. However, the RP argues that the annealing effect would be temporary and since the CHF would not be exceeded in fuel with irradiation sufficient for high fission-gas pressure, the fuel would remain operable.
149. The issues detailed above are not fundamental to the acceptability of the ABWR and I judge that suitable technical arguments and operating limits will be derived. I anticipate examining the arguments further in Step 3.

4.4 Spent Fuel Storage

150. The fundamental safety requirement for spent fuel identified by IAEA is that doses to persons as a consequence of the storage of spent fuel, are required to be kept within specified dose limits and radiation protection is required to be optimized within dose constraints.
151. In the context of fuel, this leads to the requirement that, the effectiveness of the fuel clad as a passive barrier to the release of fission products be considered (Ref. 3).
152. In particular, I note the IAEA guidance that in the design of spent fuel storage and handling systems, the spent fuel clad should be protected during storage against degradation in normal operational states and accident conditions and, later, during retrieval of the spent fuel. Furthermore, containment should be ensured by at least two independent static barriers (Ref. 4).
153. While these issues will principally be addressed in the Radwaste technical area (Ref. 9), it will be necessary to consider the fuel operating limits during irradiation to ensure that post-operation storage is a consideration in fuel and clad design. In addition to this, it is necessary to consider the resilience of the facility to reasonably foreseeable faults.

4.4.1 The Spent Fuel Pond

154. Immediately after discharge, the fuel will be stored under water due to the level of decay heat that needs to be removed. The main safety issues to consider are the maintenance of a safe subcritical configuration and the adequacy of cooling.
155. The RP reports a criticality safety target of maintaining a reactivity margin of 5% (Ref. 8). This is in accordance with industry practice and I judge it to be acceptable.
156. I asked the RP whether the case considered the potential for criticality resulting from fuel handling accidents. While it is not specifically addressed in the safety case at present, the RP argues that, should a fuel assembly be dropped onto the top of the fuel racks, the handles on the top nozzle will ensure that sufficient separation is maintained to prevent significant neutronic coupling with the fuel in the racks. I will expect this to be justified during Step 3.
157. Ref. 8 acknowledges the need to take measures to avoid pond boiling to minimise the dispersal of activated crud initiated on the fuel. This is welcome and I am aware of measures taken to reduce crud formation. I will expect this to be justified during Step 3.
158. Measures to control pool water chemistry are outside the scope of this assessment and are addressed in Ref. 10.

4.4.2 Interim Fuel Storage

159. After the fuel decay heat has reduced to an acceptable level, the RP proposes to transfer the spent fuel from the pond to interim storage outside the reactor building. This is welcome as a hazard reduction measure.
160. Two potential storage methods are under consideration: Dry cask storage under an inert gas; and wet storage in an auxiliary fuel pond (Ref. 17). The exact details do not need to be finalised at this point, but the RP wishes to demonstrate that the fuel could be stored by either method. In order to demonstrate this, it is necessary to identify degradation mechanisms for fuel in storage and to define achievable limits and conditions of operation. Assuming these limits and conditions for storage, it will be necessary to demonstrate that they are compatible with the limits for fuel operating in the reactor (e.g. discharge irradiations must be consistent).
161. In this context, I asked regulatory query RQ-ABWR-0071 on degradation mechanisms relating to dry fuel storage. I have received a draft response proposing that in dry storage, the temperature of the environment shall be controlled in normal operational occurrences to remain under 400C to maintain the fuel integrity (to avoid clad creep rupture and hydride brittle failure). Furthermore, during the fuel drying process, the clad integrity will be evaluated based on a postulated temperature history. I judge that a thermal limit of 400C limits the likelihood of clad burst as a result of creep deformation exceeding the material ductility, but in discussions, the RP acknowledged that this will not necessarily prevent clad failure by hydride-assisted cracking in the presence of radial hydride platelets (depending on the stress levels). They argued that in the event of a through-thickness chemically-assisted crack, in inert gas dry storage, there is not a fuel degradation mechanism or a mechanism for further crack growth.
162. The RP also reported that Zircaloy 2 fuel generally has low hydrogen levels at the end of its irradiation and in addition, the pure zirconium barrier layer acts as a getter for hydrogen so that, especially after slow cooling, a region of the clad is likely to exist in which hydride-assisted cracking is less likely. The RP acknowledged that this area needs further investigation.
163. The topic of dry fuel storage has been the subject of significant amount of research in recent years as plant operators relax limits on fuel pin internal pressure in order to enable higher fuel irradiations, and therefore I recognise that there will be a degree of uncertainty in any justification of clad integrity at the proposed irradiation levels. However, I expect such an argument to be made for a credible set of design limits.
164. It may be that a limited number of fuel failures result in tolerable levels of fission-product dispersal. If so, this will allow me to adopt a graded approach to assessment (within the bounds of my topical area), but the principal argument should remain a deterministic demonstration of clad integrity.
165. I also expect that the RP will make commitments on the levels of monitoring of any dry storage casks for fuel clad failure and loss of inert gas.
166. I recognise that this topic may represent a significant amount of work for the RP and therefore I will consider raise a Regulatory Observation to enable this to be effectively planned.

4.5 Analysis Methods

167. Detailed consideration of the validation of analysis methods will take place in Step 3 and 4. However, I have considered the modelling to determine whether it meets the requirements of SAP FA 18 and 24 which require that calculational methods used for the analyses should adequately represent the physical and chemical processes and take account of operational experience and advances in modelling techniques.
168. Where detailed assessment of structural analysis is required, standard industrial methods are employed. However, in the case of fuel thermal and hydraulic modelling, the complexity of the physics mandates specialist codes. Some consideration of these is given below.

4.5.1 Fuel Pin Modelling

169. Some details of the models in the fuel thermal code and the reactor physics codes are given in Ref. 7 and 11, with much more expected in topical reports to be published in August 2014. The fuel performance code appears to be constructed in accordance with modern practice, including all the models I would expect. These include:
- representation of thermal and irradiation creep;
 - self shielding of thermal neutrons within the pellet;
 - inter-linkage of pores formed at fuel grain boundaries;
 - thermal conductivity degradation with irradiation damage of the fuel pellet;
 - pellet end effects on stress; and
 - pellet cracking.
- I will examine the topical report justifying this modelling in Step 3.

4.5.2 Reactor Physics Modelling

170. The reactor physics code uses methods optimised for speed of computation. They use diffusion, rather than transport methods to generate the macroscopic cross-section data needed for the core simulator and the core power distribution is found using a composite one-group diffusion calculation; with epithermal and thermal neutrons incorporated as source terms. I need to give this detailed consideration in Step 3 to determine whether any bias or uncertainty is introduced into the analysis by modelling approximations.
171. In view of the complexity of the distribution of moderator density, fuel enrichment and the use of gadolinium to control assembly form factor, I have commissioned independent confirmatory analysis of this aspect of the analysis. This will be an assessment activity during Step 3 and probably Step 4.

4.5.3 Thermal-hydraulic Modelling

172. Thermal hydraulic modelling is based on one-dimensional, heavily empirical, representations of the fuel and core. These are substantiated by full-scale experiments on fuel assemblies.
173. Linearised frequency-domain analysis is used to demonstrate hydraulic stability.
174. The critical heat flux is evaluated based on full bundle experiments using electrical heaters. This approach is essentially based on extrapolating experimental data and is valid in so far as the model is used within its domain of validity and the body of data is well represented by the model (including allowances for uncertainty or any trend).

175. The heat flux correlation employed correlates the critical power level against local thermal hydraulic quality and the distance between the onset of boiling and the dryout position. This approach goes some way to capturing the local droplet concentration and some means of correlating the effect of the heat flux axial shape.
176. Radial effects are captured by a correction taking into account the rating of adjacent fuel pins. Both axial and radial profile effects depend on complex features of the flow such as void drift and cold-wall film thicknesses, which will vary in a complex manner. The approach is very empirical and its validity will depend on the degree of extrapolation required.
177. I will examine the arguments the RP presents on the region of validity; random and systematic error trends in Step 3.

4.6 Considerations in the Light of the Fukushima Accident

178. During the Fukushima accident, concern was raised about the integrity of the spent fuel pond in the context of very low frequency, high magnitude seismic loads. This has led to research worldwide on the potential for successful long-term cooling of the fuel in the event that the pool experienced a large breach. The RP claims to have such mitigation measures. Arguments justifying the effectiveness of these will be examined in Step 3.

4.7 Comparison with Standards, Guidance and Relevant Good Practice

179. In Section 2.2 above, I have listed the standards and criteria I have used during my GDA Step 2 assessment of the UK ABWR Fuel and Core to judge the adequacy of the preliminary safety case. My overall conclusions in this regard can be summarised as follows:
 - SAPs: The relevant SAPs are detailed in Table 1. They provide details of ONR expectations for a suitable safety case. The formal submissions to date cover some of the key safety principles adequately, but do not explicitly demonstrate compliance with all of the SAPs. This is expected to be achieved by lower-tier documentation expected for Step 3. However, ongoing dialog gives me confidence that the preconstruction safety report and associated topical reports will significantly improve the position.
 - TAGs: provide further explanation of the expectations detailed in the SAPs and while there is currently no specific TAG relevant to fuel and core, the TAG on spent fuel storage underlines the requirement to demonstrate fuel clad integrity during storage. This provided the basis for my plan to focus on this aspect during Step 3.

4.8 Interactions with Other Regulators

180. I have not undertaken any interaction with other regulators during GDA Step 2 except informal contact during international meetings. I intend to continue that informal approach at least initially during Step 3 and will attend the LWR fuel performance conference in September 2014 in Japan.
181. If opportunities to collaborate occur in the framework of the Multi-national Design Evaluation Programme (MDEP), these will be examined and if appropriate, included in my plans.

5. CONCLUSIONS AND RECOMMENDATIONS

182. The RP has provided a PSR for the UK ABWR for assessment by ONR during Step 2 of GDA. The PSR together with its supporting references, present many of the claims necessary to underpin the safety of the UK ABWR in the area of Fuel and Core.
183. During Step 2 of GDA I have conducted an assessment of the parts of the PSR and its references that are relevant to the area of Fuel and Core against the expectations of the SAPs and TAGs. From the UK ABWR assessment done so far, I conclude the following:
- The UK-ABWR design includes a fuel assembly that has demonstrated very good reliability in operation.
 - The reactor has diverse means of shutdown that meet our high-level requirements for diversity and fault tolerance.
 - The core monitoring system provides good coverage of the core, but I will need to be satisfied that it is able to detect core anomalies adequately.
 - Technical presentations on fuel modelling and the response to challenge on the topic of fuel clad integrity in postulated frequent faults have been of a high quality. This will need to be reflected in the documented safety case.
 - In the area of interim dry storage, I look forward to further detailed technical engagement to provide a substantiated set of design rules for the fuel.
 - The safety case documentation is not complete at present, but I have been encouraged by the very substantial amount of progress apparent during Step 2.
 - During Step 3, I will focus on the treatment of uncertainty in substantiation of safety margins and the quantification and justification of operating limits and rules for components of the fuel and core system generally (including non-fuel components).

5.1 Conclusions

184. I have concluded that the BWR fuel and core has achieved a high level of reliability in normal operation and The RP is likely to be able to define a set of design criteria and operating limits for use to demonstrate no fuel failures in frequent faults and acceptable releases to the environment in infrequent faults.
185. These limits include definition of bounding assumptions on core performance which underpin the demonstration of acceptable safety margins in postulated faults. Arguments needed to justify these limits and the fuel modelling required, will be examined in Step 3.
186. Overall, I see no reason, on Fuel and Core grounds, why the UK ABWR should not proceed to Step 3 of the GDA process and I consider the progress towards developing a suitable safety case satisfactory.

5.2 Recommendations

187. My recommendations are as follows:
- Recommendation 1: The UK ABWR should proceed to Step 3 of the GDA process.
 - Recommendation 2: All the items identified in Step 2 as important to be followed up should be included in ONR's GDA Step 3 Assessment Plan for the UK ABWR Fuel and Core. These are summarised in Table 3.

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- 14 *BWR Corrosion Experience of NSF Channels*, LWR Fuel Performance Meeting/TopFuel 2013, September 2013. www.ans.org/meetings/file/421.
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- 16 *Light Water Reactors Fuel Assembly Mechanical Design Evaluation*, ANSI/ANS 57.5 American Nuclear Society, 1981.
- 17 *Spent Fuel Interim Storage for UKABWR*, GA91-9101-0001-00028, March 2014, TRIM Ref. 2014/109569.
- 18 *Domestic Licensing of Production and Utilization Facilities 10CFR PART 50* www.nrc.gov/reading-rm/doc-collections/cfr/part050
- 19 *Generic PCSR Chapter 8: Reactor Core*, GA10-9101-0100-08000, UE-GD-0144, May 2014, TRIM Ref. 2014/209727.
- 20 *UK ABWR Document Tracking Sheets*. Updated versions submitted to the Joint Programme Office (JPO) throughout GDA Step 2. June 2014, TRIM Ref. 2014/214358.
- 21 *ABWR GDA – Reactor Chemistry Workshop*, XM-GD-A069, January 2014 TRIM Ref. 2014/21596.

Table 1
Relevant Safety Assessment Principles Considered During the Assessment

SAP No and Title	Description	Interpretation	Comment
EKP-	Engineering Principles Key	Requirements for defence in depth	The expectation is that the fuel will provide a robust barrier to release of activity subject to a graded approach.
ERL-	Reliability Claims	Measures to achieve reliability	
EAD-	Ageing and Degradation and	Demonstration of a safe working life and margins to safety limits throughout life.	
ERC-	Reactor Core	Measures to ensure safe operation and shutdown	
EMT-	Maintenance, inspection and testing and	Identification of maintenance and testing requirements.	This is generally not addressed in detail at a claims level,
FA-	Validity of Data and Methods	Theoretical models and calculation methods	This is discussed in Step 2 in the context of identifying a scope for Step 3.

Table 2
Relevant WENRA References to be Considered During the Fuel and Core Step 2 Assessment

Reference	Title / Description
3.1	The plant shall be able to fulfil the following fundamental safety functions: - control of reactivity, - removal of heat from the core and - confinement of radioactive material.
7.2	Criteria for protection of the fuel pin integrity, including fuel temperature, DNB, and clad temperature, shall be specified. In addition, criteria shall be specified for the maximum allowable fuel damage during any design basis event.
8.7	The impact of uncertainties, which in specific cases are of importance for the results, shall be addressed in the analysis of design basis events.
9.5	The means for shutting down the reactor shall consist of at least two diverse systems.
9.6	At least one of the two systems shall, on its own, be capable of quickly rendering the nuclear reactor sub critical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure.
5.2	Safety limits shall be established using a conservative approach to take uncertainties in the safety analyses into account.
2.1	The licensee shall assess structures, systems and components important to safety taking into account of relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.

Table 3**Items to follow up in Step 3**

Topic	Description
Fuel System	control limits for corrosion and hydriding and an appropriate programme of surveillance to support the fuel performance claims
Fuel System	Management of failed fuel
Fuel System	Safety case against PCI failure
Fuel System	SCC in non-fuel components (e.g. control blades)
Fuel System	ALARP case for mitigating the effect of channel bow
Fuel System	Crud and oxidation design substantiation documents
Fuel System	Interim dry fuel storage design limits
Fuel System	Snag loads on upper tie plate
Control Rod	Irradiation limits and ALARP justification
Control Rod	Tolerability of defects and SCC
Core Design	Specification of rod worth limitation system
Core Design	Treatment of analysis uncertainties
Core Design	Kinetic response to pressure transients
Core Design	Adequacy of core monitoring
Core Design	Fuel misloading safety case
Fuel Thermal hydraulics	Substantiation of uncertainties in MCPR (in particular, effect of axial and radial power shape)
Fuel Thermal hydraulics	Tolerability of short periods of dryout
Analysis Methods	Appropriate justification of models and correlations employed, including uncertainty and bias allowances
Analysis Methods	Suitability of fuel performance modelling
Fukushima Response	Effectiveness of fuel pond cooling in severe accidents