Health and Safety Executive

NUCLEAR DIRECTORATE

GENERIC DESIGN ASSESSMENT – NEW CIVIL REACTOR BUILD

STEP 3 FUEL DESIGN ASSESSMENT OF THE WESTINGHOUSE AP1000

DIVISION 6 ASSESSMENT REPORT NO. AR 09/040-P

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EXECUTIVE SUMMARY

This report for *Fuel Design* presents the results of ND's Step 3 assessment of the thermal and mechanical design of the fuel assembly. It provides an overview of the safety case presented in the Pre-Construction Safety Report (PCSR); the standards and criteria adopted in the assessment; and an assessment of the claims, and arguments provided within the safety case.

For Step 3 of Generic Design Assessment, I undertook an assessment, on a sampling basis, primarily directed at the supporting arguments. On the topic of *Fuel Design* this included consideration of the need to demonstrate compliance with the As Low as Reasonably Practicable (ALARP) principle and the requirement to follow international good practice.

The fuel design proposed for loading into the AP1000 reactor is a development of the existing assembly supplied by Westinghouse for irradiation in Pressurised Water Reactors worldwide. The design substantiation identifies a set of design requirements for the fuel assembly and derives a set of design criteria for analysis. These design criteria are very similar to the criteria used for Sizewell B and the analysis approach is similar.

My assessment sample included a review of the design criteria against which the fuel integrity is assessed. This is important for the fuel design because these parameters determine the boundary of safe operation for the fuel and are the basis for judging the success of fault analysis.

My assessment in the *Fuel Design* area commenced part-way through Step 3 and so it has been limited in extent, concentrating on areas where PWR operating experience has highlighted fuel performance shortfalls.

I conclude that the RP has provided a wide ranging safety analysis in the *Fuel Design* topic area and that the substantiation of claims and arguments for the scope assessed is generally adequate for GDA Step 3 but with certain shortfalls detailed below. Overall on fuel design grounds I see no reason why AP1000 should not proceed to Step 4.

- I believe that further work is required and additional information needs to be provided on PCI, crud, CHF and high-temperature fuel deformation.
- The control of reactor coolant chemistry has a significant effect on fuel performance in normal operation; especially on the likely levels of crud deposited on the fuel. This aspect of the design has yet to be finalised.
- The criteria for peak fuel enthalpy in faults needs to be updated to reflect modern practice for fuel at moderate irradiation levels, and the clad stress limit needs to be reduced to better reflect the effect of rapid power changes on the likelihood of clad failure.
- On the documentary level, a number of additional criteria are implicit in the design substantiation and would benefit being included formally. These include the peak fuel corrosion and the peak irradiation levels.
- The long term storage of the spent fuel in the period before final disposal is currently an active area of research and this will be assessed in Step 4 as information becomes available.

LIST OF ABBREVIATIONS

ALARP	As Low As Reasonably Practicable
ASME	American Society of Mechanical Engineers
BMS	(Nuclear Directorate) Business Management System
CHF	Critical Heat Flux (for departure from nucleate boiling)
EPRI	Electrical Power Research Institute
GDA	Generic Design Assessment
HSE	The Health and Safety Executive
IAEA	International Atomic Energy Agency
LOCA	Loss-of-coolant Accident
ND	The (HSE) Nuclear Directorate
PWR	Pressurised Water Reactor
RAPFE	Radial-averaged Peak Fuel Enthalpy
PCSR	Pre-construction Safety Report
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query
PCI	Pellet Clad Interaction
PIE	Post Irradiation Examination
RCCA	Rod Control Cluster Assembly
RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
RP	Requesting Party
SAP	Safety Assessment Principle
WEC	Westinghouse Electric Company LLC

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1 INTRODUCTION

1 My report presents the findings of the fuel design assessment of the Westinghouse AP1000 Pre-Construction Safety Report (PCSR) (Ref. 1) undertaken as part of Step 3 of the HSE Generic Design Assessment (GDA) process. My assessment has been undertaken in line with the requirements of the Business Management System document AST/001 (Ref. 2) and it's associated guidance document G/AST/001 (Ref. 3). AST/001 sets down the process of assessment within the Nuclear Directorate (ND) and explains the process associated with sampling of safety case documentation. The Safety Assessment Principles (SAPs) (Ref. 4) have been used as the basis for the assessment of the fuel design as has Section 6 of the relevant International Atomic Energy Agency (IAEA) standard NS-G-1.12 (Ref. 5). These standards require that fuel in a nuclear power plant can withstand normal operation and anticipated operational occurrences such as frequent faults and that releases of fission products be limited in all design-basis faults. This must be demonstrated in safety assessments and documented in a coherent set of safety case documentation.

2 Ultimately, the goal of assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety case. My report gives an initial view based on a limited sampling.

3 The objective of the Step 3 assessment is set down in Ref. 6. A review of the safety aspects of the proposed reactor design has been conducted by examining the claims and arguments made in the preliminary PCSR.

4 My assessment has not considered, in depth, core components inserted into the fuel (such as control rods and neutron sources) except to note that these components appear to be conventional in design and I do not expect their design to introduce any significant new safety issues.

5 Assessment during Step 4 will address the adequacy of the evidence supporting the claims and arguments assessed within Step 3 and will extend the scope to consideration of core internal components.

2 NUCLEAR DIRECTORATE'S ASSESSMENT

2.1 Requesting Party's Safety Case

6 The development of the proposed fuel assembly design has been an incremental process over many years and in recent years the development effort has been focused on measures necessary to enable increased fuel irradiation and also on defect reduction to meet demanding industry reliability targets. While a number of design changes are proposed for AP1000, much of the detail remains unchanged from current fuel used in existing reactors. The analysis of the fuel is supported by a programme of detailed post-irradiation examination.

7 The design of fuel assemblies is based on a set of functional requirements. These are defined in Chapter 4 of the Generic Design Report (Ref. 7). The requirements are translated into a set of design criteria, which form the basis for defining the safety case envelope. Some of these criteria represent limiting conditions in which damage to the fuel is avoided, and others indicate the region in which the relevant component can continue to meet a particular safety function for the duration of the postulated fault despite some damage.

8 Analysis is performed for normal operation and anticipated transients to demonstrate that the fuel will not breach those design criteria intended to ensure continued integrity of the fuel. For less frequent fault sequences limited fuel damage may be accepted, but analysis is carried out to demonstrate that the design will not result in a significant radiological hazard.

9 Broadly speaking, the fuel is qualified to perform without significant damage in faults with a return frequency of more than once in one hundred years and also to contain fission products with sufficient confidence to allow the plant to reach its risk requirements.

2.2 Standards and Criteria

10 The safety assessment principles used to assess the design are detailed in Ref. 4 and the subset considered relevant to faults studies and fuel are identified in Ref. 6. Those of particular relevance to fuel are found in Annex 1, together with a brief general comment on the compliance achieved by the safety case presented at Step 3.

11 One significant shortfall against UK expectations is that the case demonstrates fuel integrity for faults with a return frequency of up to once in one hundred years, while UK practice is to demonstrate compliance with similar criteria for faults with a return frequency of up to once in one thousand years. This is principally a fault-study issue and is not addressed in this report but it could have implications for fuel design.

12 Westinghouse have carried out analysis of the structural components of the fuel against American Society of Mechanical Engineers (ASME) Code Section III, which sets limits on local and membrane stresses to prevent static collapse of structures and exhaustion of material ductility. This is an established practice within the nuclear industry and elsewhere. However, I note that these limits do not necessarily prevent failure by mechanisms such as corrosion-induced embrittlement or the interaction of corrosion with crack growth (stress-corrosion cracking). This has mostly been addressed on a case-by-case basis in the absence of a generic approach, but a shortfall has been identified (see Section 2.3.1.4).

13 Detailed review of the lower-level documentation has not been carried out at this stage, but I believe that the practices employed in developing the safety case are similar to those adopted for Sizewell B with the notable exceptions considered in the assessment detailed below.

2.3 Nuclear Directorate Assessment

14 Consideration of the design criteria is presented below, followed by assessment of the key ALARP decisions made in the fuel optimisation. My examination of the safety case has to date mostly been confined to consideration of the results of Westinghouse analysis and the claims made. Detailed examination of the evidence and the analysis methods will continue in Step 4 of the assessment.

2.3.1 Design Criteria

15 The design requirements, and the criteria derived from them, are similar to those currently employed in Sizewell B and the safety case makes a systematic attempt to meet the Engineering Key Principles (EKP.2 and EKP.3) of demonstrating fault tolerance and defence in depth.

16 I have compared the design criteria against those of Sizewell B. The values are largely consistent and reflect established practice. However, I note the following points in relation to AP1000:

- The criteria limiting clad oxidation thickness is omitted.
- The criteria limiting clad hydrogen uptake is omitted.
- No limit is placed on fuel assembly irradiation.

- The limit on Radial-averaged Peak Fuel Enthalpy (RAPFE) deposited in the fuel during faults has not been changed in line with proposals derived from recent research.
- The limit on clad stress is set at the yield stress - as determined by the 0.2% proof stress - and does not account for the possibility of stress-corrosion cracking.

17 These issues are discussed in turn below.

2.3.1.1 Clad Corrosion Criteria

18 Limits are placed on fuel cladding oxidation thickness and clad hydride uptake to ensure that the fuel cladding ductility remains satisfactory. This allows other design criteria to be achieved. Westinghouse proposes to restrict the oxidation thickness and the radial mean hydride levels to traditional values. (Ref. 8). These values are supported by data derived from post-irradiation examination of the cladding material. Incorporation of these values into the set of formal design criteria would improve transparency and provide increased confidence of compliance. Consideration should be given to means of verifying compliance by fuel inspection during refuelling outages in line with Electric Power Research Institute (EPRI) recommendations.

19 The oxidation and hydride limits (together with the limit on rod internal pressure) affect not only safe operation within the reactor, but potentially also long-term storage of spent fuel. I will require a justification of the limits in the context of Westinghouse’s spent fuel storage plans in Step 4.

2.3.1.2 Fuel Irradiation Criterion

20 The limit normally placed on irradiation defines the boundary of qualification for analysis computer codes and also limits the requirement for consideration of material changes associated with fuel transmutation. The irradiation limit currently proposed is 62 MWd/kgU (pin mean). This level is within the bounds of current operation worldwide and I judge that it is probably low enough for issues relating to fuel transmutation to be tolerable. However, I will consider this further during Step 4. I feel that the proposed limit could benefit from being treated as a design criterion and a limiting condition of operation within Technical Specifications (as is the case at Sizewell B).

2.3.1.3 Peak Fuel Enthalpy

21 The proposed limit on Radial-averaged Peak Fuel Enthalpy (RAPFE) reflects a concern to prevent fuel cladding melt in rapid power transients. This concern has been augmented in recent years by a desire to prevent fragmentation of high-burnup fuel in the event of early cladding failure. Westinghouse has indicated that they propose to revise the RAPFE criterion to reflect this practice as recommended by the subject matter experts at the Electrical Power Research Institute (EPRI). I will review material provided during the Stage 4 assessment.

2.3.1.4 Clad Stress

22 The proposed clad stress limit is protective against exhaustion of clad ductility in normal operation and transients, but is not necessarily protective against stress-corrosion cracking (Appendix II of Ref. 9). Westinghouse has acknowledged this, but believes that the conservatism in their current analysis method prevents a robust demonstration of compliance with the requirement of fault tolerance (engineering principle EKP.2). This would require either significant conservatism be removed from the analysis or that

measures be put in place to mitigate power-distribution transients. I will raise this as a Regulatory Observation (RO). It is possible that additional operational constraints and protection measures may be required. Westinghouse will be required to provide proposals in Step 4.

2.3.2 ALARP Measures Taken To Optimise Fuel

23 The proposed fuel is the result of a systematic programme of optimisation aimed at eliminating defects and increasing safety margins. Operation experience demonstrates that progress to date has resulted in a robust assembly design with markedly reduced in-service defects compared with the first V5H fuel assemblies loaded into Sizewell B. The notable changes are considered below.

2.3.2.1 Fuel Pellet Design

24 The fuel pellet will be a standard Westinghouse product. It will be manufactured to a similar density to those currently loaded in Sizewell B and will be irradiated up to a maximum pin mean burnup of 62 MWd/kgU. While this is higher than current UK experience, it is still lower than irradiations being routinely achieved in mainland Europe and an extensive body of post-irradiation inspection data is available to support the qualification of the fuel to this level of irradiation.

25 At irradiations in the region of 50-60 MWd/kgU, fuel starts to undergo physical changes in its outer rim and accumulation of fission-product inclusions affects the crystal structure of the uranium oxide. Existing experimental evidence suggests that this effect is largely benign until higher levels of irradiation are reached and therefore the proposed limit is reasonable based on the current evidence, but I will review recent programmes of research in Step 4.

26 The reactivity of the fresh fuel will be constrained on initial load by the use of a coating of Zirconium Diboride. This is a well-established practice within the USA, but is new to the UK. The uniformity of the boride coating has been a performance concern in the past, but enhanced manufacturing controls have been introduced. The manufacturing limits need to be reflected in safety case uncertainty allowances. I will review this in Step 4.

27 The pellet material is potentially vulnerable to damage during manufacture, leading to increased local stresses on the cladding. Westinghouse has examined this effect and their stress analysis method takes account of permissible defects. The standard against which the fuel is manufactured has been tightened and an automatic process of fuel inspection is under development. These mitigation measures are commendable and are useful in providing margin to the limits required in Section 2.3.1.4 above.

28 Measures are also in place to limit the scratching of the fuel cladding during manufacture including inspection against traditional limits, which based on operational experience appear adequate.

2.3.2.2 Fuel Pin Plenum Design

29 The fuel pin will include a conventional plenum at the top to accommodate fission-product gas released from the fuel and to house the spring which holds the fuel pellets in place. However, this is augmented by a lower plenum. The lower plenum is engineered in the form of a metal washer supporting the bottom pellet and a tube supporting the washer. I will examine this arrangement further in Step 4.

2.3.2.3 Cladding Selection

30 The Zirlo fuel cladding has improved performance over the traditional Zircaloy by reducing the level of cladding oxidation and the associated hydride embrittlement of the cladding. The material has now been widely used and data from operating plants supports its use. The Post Irradiation Examination (PIE) data presented by Westinghouse showed a stable oxide film with little tendency to spall even at high levels of oxidation and cladding ductility is retained well beyond the irradiation levels envisaged. During Step 4 I will confirm that this is satisfactorily documented within the safety case.

31 I have reviewed the cladding growth data and statistical analysis of uncertainty has been reviewed by the US Nuclear Regulatory Commission (US NRC). They found the data and analysis to be acceptable - giving me confidence that excessive distortion of the fuel is unlikely (Ref. 8). However, I do not consider that this removes the need for suitable fuel surveillance which will be considered further in Step 4.

32 There is an alternative material that could have been used for the cladding called Optimised Zirlo. This is relatively new and relies more on Niobium (Nb) precipitation hardening for its strength. This could give better overall corrosion properties. However, more data would be beneficial before loading this material into new reactor systems. I think that the current approach using Zirlo cladding is sensible at this time but this conclusion is subject to the matter below.

33 Cladding strain when overheated in large loss-of-coolant accidents (LOCA) is currently analysed by Westinghouse based on correlations relating diametric strain at failure to clad temperature. I do not consider these correlations sufficiently general to be applied to new cladding materials, pin designs or changes in the general form of the LOCA thermal transient. A more detailed analysis of the material strain distribution in a particular event is considered necessary to give confidence that the experimental data obtained for Zircaloy remains applicable. I will explore this further as part of Step 4.

2.3.2.4 Spacer Grid Design

34 The proposed spacer grid design has shown itself able to withstand hydrodynamic forces in service at coolant flow rates in excess of those envisaged. Furthermore, the introduction of intermediate mixing vanes will provide additional structural stiffness for the assembly. This gives me some confidence that fretting will not be a significant issue for the design, although this does depend on the as-built plant and fuel assemblies.

35 Consideration of the effect of dynamic forces associated with rapid depressurisation of the primary circuit has been limited to the case of a fracture of the surge line. This is based on the primary-circuit break exclusion argument. This aspect of the case needs to be analysed by Westinghouse if successful mitigation of the 2A size large LOCA is to be claimed as a success in the Probabilistic Safety Analysis (PSA).

36 The irradiation growth of the spacer grid is modelled by a general empirical correlation which allows for two components: zirconium-hydride precipitation and irradiation creep. The data is consistent with the correlation and the reduced corrosion associated with the change to Zirlo is likely to provide a satisfactory margin to the available space within the current irradiation limit consequently I judge this to be satisfactory.

37 The design of the spacer grid edge has been modified to reduce the likelihood of damage during fuel handling, but it retains a geometry designed to enhance swirl and turbulent mixing. Observations of oxidation and crud levels in the region down-stream do not indicate any anomalies. I therefore judge that this design is less likely than others to experience anomalous heat transfer in the down-stream region and have chosen not to focus my review on this issue as the position is likely to be satisfactory.

38 The margin to Critical Heat Flux (CHF) that the coolant flow can safely remove from the fuel pins without boiling heat transfer failing has been characterised for the design and the physical phenomena, which could introduce uncertainty into the quantified limit have been studied and accounted for in the analysis. This study includes certain parameters which are incorporated in a statistical analysis of uncertainty and other phenomenon which are allowed for explicitly in the analysis. However, I am still considering the adequacy of these allowances.

39 The effects of irradiation-induced distortion are specifically allowed for by a rod-bow allowance, which takes account of the possibility that the proximity of neighbouring rods may influence the CHF. Westinghouse claims that this allowance will also accommodate power increases caused by pins bowing apart – at least up to the levels expected. This claim is supported by recent developments in analysis methods, but the claim needs to be substantiated by documented analysis as part of the safety case and formal surveillances are required to ensure that the plant continues to operate within the bounds of its safety case.

40 The effect of modest levels of crud has been incorporated into CHF analysis, but this has not been linked to a surveillance on crud levels and the analysis has omitted to consider the likelihood that crud will deposit preferentially on the highest-rated fuel pins. The control of reactor coolant chemistry has a significant effect on fuel performance in normal operation; especially on the likely levels of crud deposited on the fuel. This aspect of the design has yet to be finalised.

41 Westinghouse claims that the performance of the assembly edge in CHF tests is bounded by that of the central mixing vanes. This data needs to be formally reported and incorporated into the safety case. The report should include consideration of the effect of closing the gap between adjacent assemblies.

42 The issues identified in paras 35, 38, 39, 40 and 41 are the subject of current discussions and may result in Regulatory Observations (RO) but will require resolution in Step 4.

2.3.2.5 Bottom Nozzle Design

43 The bottom nozzle is manufactured from stainless steels with resistance to stress-corrosion cracking. It is a variant of current designs and I feel reasonably content with it in concept.

44 The assembly is optimised to reduce the flow resistance by adding a small chamfer on the holes in the perforated plate. I do not expect this change to be risk-significant. The feet of the nozzle have been shortened as a result of the addition of the pin lower plenum. I will examine this modification in Step 4 with the assistance of structural integrity experts.

45 In combination with the bottom grid, the bottom nozzle provides an efficient trap for debris. This, together with the use of surface hardening of the bottom end of the fuel pins, has been shown to be effective mitigation of the effects of debris entering the core and I judge this element of design to be acceptable.

2.3.2.6 Top Nozzle Design

46 The top nozzle has undergone significant modification to reduce the risk of stress-corrosion cracking of the bolts securing the assembly hold-down springs. This is achieved by pegging the springs directly into a slot in the body of the nozzle – removing the need for any pre-tensioned components. I think that this approach is reasonable, but will examine it in more detail in Step 4 with the assistance of structural integrity experts.

2.3.3 Regulatory Observations

47 No formal Regulatory Observations (RO) have been issued to date. However, I will raise the current shortfall in the analysis of clad stress (Section 2.3.1.4) as a Regulatory Observation.

48 Assessment against a revised RAPFE limit has not been raised because this work is already in hand. This may also apply to consideration of the effects of fuel crud and assembly distortion.

2.3.4 Plans for Step 4

49 In the assessment for Step 3, I have focused on identification of an appropriate and well-defined boundary to the safety case. In Step 4 I will extend the scope of assessment to Rod Control Cluster Assemblies (RCCA) and other inserted components and will examine the evidence presented to support the boundary definition and to ensure compliance. This will include ensuring that the documentation of the evidence and arguments is sufficient to constitute a satisfactory safety case. Specific areas identified for detailed consideration are given below:

- Proposals for demonstrating no clad failures due to thermal stress in postulated frequent faults will be considered.
- Westinghouse has indicated that they propose to revise the Radial-averaged Peak Fuel Enthalpy (RAPFE) criterion to reflect good practice - as recommended by the subject matter experts at EPRI. The basis of this revised criterion will be reviewed.
- The case for operation with surface crud on the fuel will be examined. The implications for CHF will be assessed and the proposed measures for surveillance considered.
- Assessment of the effect of changes to fuel design and cladding material on the arguments made for preservation of coolable fuel geometry in large LOCA faults will continue. It is anticipated that this will include detailed modelling of the deformation of the fuel assembly in a postulated fault.
- The performance of the edge of the spacer in CHF tests will be further examined.
- The design substantiation of novel components - such as the fuel lower plenum - will be reviewed in greater detail.
- Longer term safety of the fuel following discharge from the reactor building into the onsite storage facility will be addressed.

3 CONCLUSIONS AND RECOMMENDATIONS

50 Westinghouse has taken a series of measures to improve the quality of their fuel in recent years and based on operational experience data, have been rewarded by improved performance.

51 I judge that the safety case is presented systematically in the fuel area and with some additions and reservations noted above, it should have the elements needed for an acceptable fuel safety case.

52 Emergent technical issues relating to the fuel have been addressed proactively. However, in some cases this is not complete or needs to be reflected in the safety case (e.g. RAPFE criteria).

53 I have found no significant shortcomings apparent in the design of the fuel assembly at this stage of the assessment, although additional operational constraints and protection measures may be required.

54 I am of the opinion that some surveillance on fuel condition may be formally required to confirm operation consistent with safety limits.

55 The control of reactor coolant chemistry has a significant effect on fuel performance in normal operation; especially on the likely levels of crud deposited on the fuel. This aspect of the design has yet to be finalised.

56 One significant shortfall against UK expectations is that the case demonstrates fuel integrity for faults with a return frequency of up to once in one hundred years, while UK practice is to demonstrate compliance with similar criteria for faults with a return frequency up to once in one thousand years. This is principally a fault-study issue and is not addressed in this report but it could have implications for fuel design.

57 Overall on fuel design grounds I see no reason why AP1000 should not proceed to Step 4.

4 REFERENCES

1 *AP1000 Pre-construction Safety Report*. UKP-GW-GL-732, Revision 1, Westinghouse Electric Company LLC, March 2009.

2 *ND BMS, Assessment Process.* AST/001, Issue 2, HSE, February 2003.

3 *ND BMS, Guide: Assessment Process*. G/AST/001, Issue 2, HSE, February 2003.

4 *Safety Assessment Principles for Nuclear Facilities*. 2006 Edition, Revision 1, HSE, January 2008.

5 *Design of the Reactor Core for Nuclear Power Plants*. IAEA Safety Guide No. NS-G-1.12, International Atomic Energy Agency (IAEA), Vienna, 2005.

6 *Generic Design Acceptance. Fault Analysis and Fuel Assessment Plan for Step 3.* ND Division 6 Project Initiation Document PID 09/040-P, Issue 01, July 2009.
TRIM Ref. 2009/178884.

7 *AP1000 European Design Control Document*. EPP-GW-GL-731, Revision 0, Westinghouse Electric Company LLC, 12 November 2008.

8 *VANTAGE+ Fuel Assembly Reference Core Report*. WCAP-12610-P-A (P), Westinghouse Electric Company LLC, April 1995.

9 *Safety of Nuclear Power Plants: Design Requirements*. IAEA Safety Series No. NS-R-1, International Atomic Energy Agency (IAEA), Vienna, 2000.

Annex 1 – Fuel Design - Summary of Assessment against HSE-ND Safety Assessment Principles

SAP Number	Assessment Topic / SAP Title	Assessment
EKP	**Key engineering**	
EKP.2	Fault tolerance	The safety case demonstrates components are resistant to faults up to a frequency of 1 x 10^{-2} per yr. This compares to a UK target of 1 x 10^{-3} per yr. The exception is clad stress for which a satisfactory demonstration is not yet available.
EKP.3	Defence in depth	At Level 1 Considerable effort has been applied to prevent failures by design and safety margins have been improved compared to earlier designs. At level 2 Some design constraints need to be reflected in technical Specifications and Surveillances.
FA –	**Design basis analysis**	
FA.4	Fault tolerance	The design-basis analysis is systematically integrated into the fuel design process by the use of a set of design criteria as fault acceptance criteria.
FA.9	Further use of DBA	The faults define the limiting conditions for operation of the fuel via the selection of design criteria. The application of these criteria is discussed in detail in the body of this report.
FA -	**Theoretical Models**	
FA.17	Theoretical models	Theoretical models should be an adequate representation. A detailed assessment of the models employed is planned for Step 4. However, I note that the fuel pin model is simplistic in omitting the axial variation of the pellet radial strain in power transients.
FA.18	Calculation methods	Validation and treatment of uncertainties is required. This will be addressed in more detail in Step 4, but sampling has indicated sound treatment of uncertainties supported by experimental and in-service data. The methods have also been reviewed by the USNRC as part of its assessment of Reference 10 and this adds confidence.
FA.19	Use of data	This will be addressed in more detail in Step 4, but sampling has indicated clearly defined limits of application of modelling supported by data. The data has also been reviewed by the US NRC as part of its assessment of Reference 10 and this

SAP Number	Assessment Topic / SAP Title	Assessment
		adds confidence.
FA.20	Computer models	Satisfactory controls on the development of computer modelling are required. This will be addressed in Step 4.
FA.21	Documentation	This will be addressed in more detail in Step 4, but sampling has indicated generally satisfactory documentation within the Document Control Document, but not all supporting material is present in the formal safety case reference trail.
FA.22	Sensitivity studies	This will be addressed in Step 4