Health and Safety Executive

NUCLEAR DIRECTORATE

GENERIC DESIGN ASSESSMENT – NEW CIVIL REACTOR BUILD

STEP 3 FUEL DESIGN ASSESSMENT OF THE EDF AND AREVA UK EPR

DIVISION 6 ASSESSMENT REPORT NO. AR 09/041-P

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EXECUTIVE SUMMARY

This report presents the findings for *Fuel Design,* of an assessment of the EDF and AREVA Pre-Construction Safety Report (PCSR) for the UK EPR undertaken as part of the Step 3 of the HSE Generic Design Assessment (GDA) process. It provides an overview of the safety case presented in the 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, Requesting Parties provided a PCSR plus principal supporting references. I undertook an assessment, on a sampling basis, by analysis of the arguments presented. On the topic of *Fuel Design* this includes consideration of the need to demonstrate compliance with the ALARP principle and to follow international good practice.

Safety arguments set out in the PCSR include that the fuel will:

- retain its integrity in normal operation and anticipated transients; and
- maintain a coolable geometry such that radiological releases from the fuel are limited in faults of lower frequency (within the scope of the design basis).

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 a basis for judging the success of fault analysis.

My assessment in the *Fuel Design* area commenced part-way through Step 3 and has been limited in extent, concentrating on areas where PWR operating experience has highlighted fuel performance shortfalls. These include: the effects of fuel assembly irradiation growth; the formation of crud; and stress-corrosion cracking in power transients.

a) I conclude that the Requesting Parties 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. Overall on fuel design grounds I see no reason why EPR should not proceed to Step 4.

b) Particular areas where I believe that further work is required or additional information needs to be provided are:

    - prevention of fuel cladding cracks (due to thermal stress) in potential fault transients;
    - the effect of conditions in core on fuel critical heat flux and;
    - the effect of mineral deposit on the surface of the fuel pins; and
    - the safety of the long term storage of the fuel before final disposal focussing on the role of the levels of burnup.

    I will assess these issues further in Step 4.

c) At this time no potential exclusions have been identified.

d) On the documentary level, the clarity with which the fuel design criteria are defined and justified merits improvement.

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)
EA	The Environment Agency
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
PCER	Pre-construction Environment Report
PCSR	Pre-construction Safety Report
PWR	Pressurised Water Reactor
RAPFE	Radial-averaged Peak Fuel Enthalpy
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query
RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
SAP	Safety Assessment Principle
SSC	System, Structure and Component
WENRA	The Western European Nuclear Regulators' Association

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1 INTRODUCTION

1 My report presents the findings of the fuel design assessment of EDF and AREVA UK EPR 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 analysis 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. This report gives an initial view based on a limited sampling. This, and the late availability of some documents, has limited the scope and depth of my assessment. However because of increased level of resources now available by the end of Step 4 any shortfall in the Step 3 work will be covered in order to produce a meaningful GDA result.

3 The 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.

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

5 Assessment during Step 4 will be extended to include all in-core components and will address the adequacy of the evidence supporting the claims and arguments assessed within Step 3.

2 NUCLEAR DIRECTORATE'S ASSESSMENT

2.1 Requesting Party's Safety Case

6 The functional requirements of the fuel are discussed in general terms in Chapter 4 of the RP's pre-construction safety report (PCSR) (Ref 1), with the resulting design criteria discussed either in this document or relevant sections of subsidiary documents such as the fuel rod thermal design report, with some performance criteria reported as part of fault analysis.

7 A useful summary of the fuel characteristics is found in Ref. 7.

8 The fuel is designed against graded functional objectives in accordance with the fault class. Four classes exist:

- Class I and II encompass normal operating conditions and anticipated transients with a return frequency greater than once in one hundred years. In these cases, the requirement is that integrity is maintained.
- Class III events are faults with a return frequency less than once in one hundred years, but greater than once in ten thousand years. In these cases, the requirement is that only 10% of fuel rods are damaged;

- Class IV events are faults with a return frequency less than once in ten thousand years. In these cases the requirement is that an acceptable coolable geometry is maintained, which in certain cases translates to 10% cladding damage.

9 Measures taken to demonstrate compliance with these functional requirements are detailed in a set of design reports and fault studies.

10 The case does not include details of the proposed formulation of Technical Specifications defining the formal boundary of safe operation.

2.2 Standards and Criteria

11 The safety assessment principles used to assess the design are detailed in Ref. 4 and the subset considered relevant to faults studies and fuel is 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.

12 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.

13 AREVA has carried out an 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. ASME Class A criteria are used for analysis of normal operation and most transients. ASME D is used for seismic and LOCA faults. The use of ASME criteria is an established practice within the nuclear industry. 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). In cases where this could be relevant, the qualification has been augmented either by component endurance testing under aggressive environmental conditions, or by review of experience with existing fuel. This has been claimed to demonstrate that the material used is appropriate to the particular application.

2.3 Nuclear Directorate Assessment

14 Consideration of the key ALARP decisions made in the fuel optimisation is presented below, followed by my assessment of the fuel design criteria and the analysis of the main components of the assembly. My examination of the safety case has to date mostly been confined to consideration of the results of the AREVA analysis and the claims made. Detailed examination of the evidence and the analysis methods will continue in Step 4 of the assessment. The assessment assumes that the design criteria and interface parameters, on which the analysis is based, will in due course be satisfactorily reflected in Technical Specifications to ensure operational compliance.

15 I have not reviewed non-fuel core components in detail except to note that the design practices used are generally not novel.

16 On a general level, I feel that the safety case on this topic could benefit from assembling the specification and justification of design criteria (and decoupling parameters) in one place; discrete from the documents in which compliance is demonstrated. This has been done to some extent, but coverage could be more comprehensive and hence the parameters assumed could be more easily controlled. In particular, arguments justifying the values assumed need to be added to make for an adequate safety case.

17 The development of the proposed fuel assembly design has been an incremental process over many years. The AREVA HTP fuel assembly has shown sustained excellent performance in meeting demanding industry targets for reliability. Much of the proposed design is based on this fuel, but the spacer grids are taken from the AFA design used more widely in France. A discussion of the optioneering process has not been presented, but AREVA claims slightly better thermal performance for the proposed design and recent changes to the AFA assembly appear to have removed the propensity for occasional fuel cladding failures by grid-rod fretting. I note that the proposed flow conditions are not expected to be as extreme as those experienced in existing N4 plant and therefore I judge satisfactory performance should be achievable.

18 In recent years, measures have been taken to enable increased fuel irradiation. These have included the use of the M5 cladding alloy, which shows greater corrosion resistance and lower irradiation growth than the conventional Zircaloy material. This change is considered in more detail below when the cladding is assessed.

19 While a number of design changes are proposed for EPR, 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.

20 In the proposed design, the potential constraint on fuel irradiation caused by fission gas release from the fuel pellet has been relieved to some extent by the introduction of a lower plenum into the rod.

21 The debris retention capability of the fuel assembly bottom nozzle has also been optimised based on experiment.

2.3.1.1 Fuel Pellet

22 The fuel pellet design is similar to that of Sizewell B. A maximum fuel burnup for the normal fuel rods is set at 62 MWd/kgU (pin mean). This has been the US limit for many years and is also within the region of operation routinely achieved elsewhere in Europe. An extensive body of post-irradiation inspection data is available to qualify the fuel to this irradiation level.

23 My judgement is that the level proposed is low enough to limit the effects of high-burnup crystal structure formation within the fuel pellet (although these effects can not be fully discounted as negligible). I will consider this further during Step 4 in the light of results from current research programmes.

24 Gadolinium will be used as a burnable poison at the same concentration levels as Sizewell B. The burnup of Gadolinium-doped pins is limited to 49.7 MWd/kgU. The limiting power level has been set to ensure that the fuel centre temperature of doped rods is broadly bounded by the normal rods. This is not explicitly defined as a design constraint, but the practice is similar to that of Sizewell B. The use of Gadolinium in this role is judged acceptable.

25 The fuel pellet enthalpy criterion used in rapid power transients is specified as 220 cal/g, reducing to 200 cal/g for irradiated fuel (Ref. 1 Chapter 14.5.5.1.1). I could find no reference providing justification for these values which implicitly assume a high level of cladding ductility. This criterion needs to be justified taking due account of recent experiments and needs to be consistent with the limits on oxidation given in Ref. 9 in order to make for an adequate safety case.

2.3.1.2 Fuel Rod

26 The fuel rod Design Criteria are set out in Chapter 3 of Ref. 8. Many of the criteria are the same or similar to those of Sizewell B. However, a number of criteria are omitted:

- A criterion for acceptable scratches on the surface of the cladding appears to be absent. Scratches can be caused by contact with the spacer grids while loading the fuel into the assemblies and mitigation measures are required to ensure that they do not prejudice the integrity of the cladding. I will examine this in Step 4.
- No constraint on local void fraction is identified in Ref. 8, but I understand that a 5% level is used for normal operation. Such a criterion is generally considered useful to ensure that the coolant chemistry remains reducing. This helps prevent abnormal levels of oxidation. The value of 5% is claimed to be based on experience, but seems to be high when compared to the Sizewell B limit of 1%. However, I note that the EPR value is evaluated using a pessimistic core flow rate, so direct comparison is not a fair test. I will review this along with general consideration of cladding oxidation in Step 4.

27 In general, the criteria given in Ref. 8 are a subset of the criteria and decoupling parameters assumed in the safety case and the definition of a more comprehensive set is necessary for control of the design and visibility of the safety case. This is particularly true for fuel related parameters used in fault studies where retaining a holistic view can be difficult. I have come to a view on the adequacy of some of these criteria using information from the general literature rather than reports provided as part of the safety case. Clear documentation is necessary for example to avoid misconceptions developing.

28 The cladding will be manufactured in the M5 material, which has recently become AREVA's standard product and is currently loaded into Sizewell B.

29 The conventional design criteria for Zircaloy oxidation levels are used for M5 cladding. These are a peak oxide thickness of 100 µm and a local radial mean hydride level of 600 ppm. The 100 µm limit is higher than the limit used at Sizewell (80 µm) which was originally reduced to reflect the poor performance of optimised Zircaloy with spalled oxide films. The spalling behaviour of M5 is different from that of Zircaloy and although the oxide appears to be more friable, it has been observed to detach as small flakes (rather than large sections of oxide exposing the underling metal). Nevertheless, even test assemblies at extremely high levels of irradiation have not experienced oxide thicknesses approaching 100 µm to my knowledge. I have no reason to suppose that these limits are inappropriate and judge this a satisfactory position but I will review recent data as part of Step 4.

30 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. A justification of the limits in the context of EDF's spent fuel storage plans will be required in Step 4.

31 The M5 material is a 1% Nb alloy similar to the E110 alloy used for many years in VVER (Russian design for light water pressurised reactor) plant. The levels of oxidation achieved during normal operation are very low compared to Zircaloy due to changes in the morphology of the oxide and extended periods of parabolic growth of the protective oxide film. The resulting hydrogen uptake is also very low. Consequentially, ductility of the fuel cladding is maintained for the proposed fuel irradiation and beyond, making the material generally satisfactory. However, in extreme cases where significant levels of boiling take place or there is contamination of the surface, the oxidation does not always follow the normal characteristic and in these cases the cladding might have impaired performance at high levels of irradiation. It follows that a constraint on the level of boiling is required and regular surveillance of fuel is required following each core offload. I will request further detail in Step 4.

32 The satisfactory performance of the cladding depends on effective cooling of its surface and maintenance of the required coolant chemistry at the oxide-metal interface. In particular, excessive crud formation must be avoided. Since the level of crud found on

the surface must be limited for other design limits to be met, this should itself be specified as a limit and appropriate surveillances introduced on fuel immediately following a core offload.

33 AREVA claims that crud can be avoided by control of PH and suitable treatment of the primary coolant chemistry during initial phases of plant operation. A number of measures have been taken to control the risk of crud formation and AREVA claims, based on experience with N4 plants, that the risk can be averted. I, in conjunction with our Reactor Chemist, will examine this claim as part of Step 4. Particular attention will be given to consideration of the role of subcooled boiling in the chemistry argument.

34 AREVA has considered the potential effect that crud may have on the core flow distribution, but are still assessing the potential effect of increased rod surface roughness on this.

2.3.1.3 Fuel Cladding Mechanical Integrity

35 The conventional limits on clad stress and strain are mostly set at traditional values. However, the analysis against these limits is not complete. This will be further examined and a complete analysis sought in Step 4.

36 No limit is currently quoted for protection against failure by stress-corrosion cracking caused by Pellet-Cladding Interaction (PCI) in power transients. This is because the means of protection against these power transients has yet to be finalised. Assessment of this issue will commence when the information is available. This will be raised as a regulatory observation.

2.3.1.4 High-temperature Cladding Deformation

37 The cladding strain and the potential for flow blockage when fuel is overheated in large loss-of-coolant accidents (LOCA) is currently assessed by AREVA based on correlations relating diametric strain at failure to clad temperature. The requirement is to demonstrate that a coolable geometry is maintained. I do not consider the correlations employed sufficiently general to be used for new cladding materials, pin designs, or changes in the general form of the LOCA thermal transient and judge that an alternative approach is necessary.

38 Ref. 9 recommends that the correlations be revised and that analysis methods be developed to account for azimuthal variation in clad temperatures. I endorse this view, but an alternative approach has also been presented by AREVA. Based on detailed consideration of the bounding rod-pressure and rating histories, analysis has been presented to demonstrate that no fuel would be expected to fail in the transient. AREVA plans to incorporate this study into the next issue of the safety case documentation. I will explore this topic further as part of Step 4. I envisage undertaking some confirmatory calculations to satisfy myself that there is not a likelihood of a significant flow blockage in this fault.

39 In the smaller LOCA, prevention of core uncovery requires that the plant be depressurised to below 10 MPa so that medium-head injection pumps can replenish inventory loss. If a cool down of 100 K/s is assumed, certain size breaks can result in core uncovery before safety injection can occur. In these conditions, the stress and thermal environment is different from those conventionally studied and other deformation mechanisms may become significant (Ref. 10).

40 AREVA has indicated that it intends to increase the rate of depressurisation in these faults and therefore to reduce the depth and duration of any core uncovery to such a level that no cladding failures are expected. This is important to support the decision to omit

high-head safety injection pumps and I will explore the analysis further as part of Step 4. I anticipate that there will be a need for me to procure confirmatory calculations.

2.3.1.5 Bottom Nozzle Design

41 The bottom nozzle design is essentially similar to the Fuelguard bottom nozzle used at Sizewell, but with a reduced spacing of the laths compared to the standard product and with a reduced-height foot.

42 This design has good flow characteristics and testing of its ability to trap debris swept into the core with the coolant shows good results (Ref 11). The design is judged acceptable in principle.

43 The structural integrity of this component has been assessed by detailed finite-element methods. I will examine this in Step 4. with the assistance of our structural specialist.

2.3.1.6 Top Nozzle Design

44 The design is similar to Sizewell B, but partly due to the removal of the instrument tube in the centre of the assembly, the nozzle has a different pattern of flow holes. They appear to preserve a larger flow area. There are also some detailed changes to the leaf springs. I will examine this in Step 4. with the assistance of our structural specialist.

2.3.1.7 Spacer Grid Design

45 Spacer grid design has shown itself able to withstand hydrodynamic forces in service when subject to flow rates in excess of those envisaged.

46 Irradiation-induced growth of the spacer has been considered and operational data has been used to demonstrate that this can be accommodated with sufficient margin to enable normal fuel handling.

47 The grid design includes mixing vanes optimised to promote swirl and turbulent mixing. 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 (Ref 8). 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. This analysis method is still under review by AREVA.

48 Grid straps at the assembly edge are however somewhat different from the rest. At the edge, a series of tabs at the top and bottom of the peripheral grid straps are bent in towards the centre of the assembly. These provide some protection against assemblies snagging during fuel off-load, but also help enhance turbulence. Generally, the level of flow disturbance in this region would be expected to enhance the CHF and this is confirmed by experiment. However, the argument still needs to be made by EDF and AREVA for the case where distortion causes adjacent assemblies to touch.

49 More generally, the effect of irradiation-induced distortion is specifically allowed for by a rod-bow allowance in the CHF analysis which takes account of the possibility that the proximity of neighbouring rods may influence the critical heat flux. AREVA claims that this allowance will also accommodate power increases for edge pins caused by assemblies bowing apart – at least up to the levels of distortion expected. This 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.

50 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 needs to be addressed if more extreme LOCA faults are to be accepted as successfully mitigated within the scope of the Probabilistic Safety Analysis (PSA). I will examine this in Step 4.

2.3.1.8 Control-rod Guide Tubes

51 The analysis of axial growth of the fuel assembly guide tubes was demonstrated to be satisfactory by a limited set of data obtained from post-irradiation examination. This needs to be augmented by more extensive sampling, but is an acceptable short term position. The outlier in the existing data needs to be investigated and conclusions included in the safety case.

2.3.2 Regulatory Observations

52 No formal regulatory observations have been issued to date. However, shortfalls are thought to exist in the consideration of the effects of fuel crud and also in analysis of the thermal performance of the edge of the assembly. Both of these topics may warrant a formal observation.

53 An observation has not been raised for the PCI issue because work is currently ongoing.

2.3.3 Plans for Step 4

54 The assessment for Step 3 has focused on identification of an appropriate and well-defined boundary to the safety case. Step 4 will extend the scope to in-core 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 to demonstrate no clad failures due to thermal stress in postulated frequent faults;
- justification of the Radial-averaged Peak Fuel Enthalpy (RAPFE) criterion to reflect good practice;
- the case for operation with surface crud on the fuel;
- implications of crud for CHF and the proposed measures for surveillance;
- CHF performance of the edge of the spacer;
- the effect of the fuel design and cladding material on the arguments made for preservation of coolable fuel geometry in large LOCA faults (I anticipate that this will include detailed modelling of the deformation of the fuel);
- design substantiation of novel components in greater detail (including the lower plenum of the rod and changes to the nozzles);
- longer term safety of the fuel following discharge from the reactor building into the onsite storage facility;
- Justification of design criteria and interface parameters.

3 CONCLUSIONS

55 AREVA has taken a series of measures to improve the performance of its fuel in recent years and has maintained fuel quality while increasing potential levels of irradiation.

56 I judge that the safety case presented in the fuel area is generally acceptable although with the reservations detailed above, but could benefit from a more complete treatment of the justification of fuel design criteria.

57 Emergent technical issues relating to the fuel appear to have been addressed proactively. However, in some cases this needs to be documented in the safety case (e.g. Justification of design criteria).

58 I have found no significant shortcomings in the design of the fuel assembly apparent at this stage of the assessment, although additional operational constraints and protection measures may be required. One area of particular concern is the protection against cladding failure by PCI.

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

60 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 needs to be further examined, with particular reference to the effect of boiling. Further evidence is required from AREVA in this area.

61 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. I will examine this aspect further in Step 4.

4 REFERENCES

1 *UK EPR Pre-Construction Safety Report.* UK EPR-0002-132, Issue 02, EDF and AREVA, June 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, Issue 01, July 2009.
TRIM Ref. 2009/178884.

7 *AFA 3GLE fuel assembly for FA3 EPR Reactor Product Sheet.* FS1-0000096-EN-3, AREVA, July 2009.

8 *Fuel Rod Thermal / Mechanical Design Basis.* DC05549, AREVA, July 2009.

9 *A State - of – the Art Review of Past Programmes Devoted to Fuel Behaviour Under LOCA Conditions – Part One Clad Swelling and Rupture – Assembly Flow Blockage.* SEMCA-2005-313, IRSN, December 2005.

10 *A Transition Stress In The Creep Of An Alpha Phase Zirconium Alloy At High Temperature.* Scripta Metallurgica, Volume 19, Issue 11, November 1985.

11 *Fuel Assembly Mechanical Design Report.* FS1-0000099-EN-3, AREVA, July 2009.

Annex 1 – Fuel Design – Assessment against HSE Safety Assessment Principles

SAP Number	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 for PCI against either target. Some areas need further justification. These include CHF and RAPFE.
EKP.3	Defense in depth	At Level 1; Considerable effort has been applied to prevent failures by design and safety margins have been improved compared to existing fuel. At level 2 Design constraints will 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 and decoupling 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, subject to reservations discussed in the main body of the report, a superficial review has not identified any significant deficiencies.
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 generally satisfactory treatment of uncertainties supported by experimental and in-service data.
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 modeling supported by data.
FA.20	Computer models	Satisfactory controls on the development of computer modeling are required. This will be addressed in Step 4.

SAP Number	SAP Title	Assessment
FA.21	Documentation	This will be addressed in more detail in Step 4, but sampling has indicated generally satisfactory documentation, although a complete document trail with substantiation of claims and assumptions are some times absent.
FA.22	Sensitivity studies	This will be addressed in Step 4