Attachment

TECHNICAL EVALUATION REPORT OF BAW-10229P (MARK-B11 FUEL ASSEMBLY DESIGN TOPICAL REPORT)

C. E. Beyer
J. M. Cuta

October 1999

Prepared for
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
under Contract DE-ACO6-76RLO 1830
NRC JCN J2435

Pacific Northwest National Laboratory
Richland, Washington 99352

LIST OF ACRONYMS

AOO	- Anticipated Operational Occurrence
ASME	- American Society of Mechanical Engineers
CHF	- Critical Heat Flux
DNB	- Departure from Nucleate Boiling
DNBR	- Departure from Nucleate Boiling Ratio
ECCS	- Emergency Core Cooling System
FCF	- Framatome Cogema Fuels
GDC	- General Design Criterion
LOCA	- Loss of Coolant Accident
NRC	- U.S. Nuclear Regulatory Commission
PCI	- Pellet Cladding Interaction
PCT	- Peak Cladding Temperature
PNNL	- Pacific Northwest National Laboratory
RIA	- Reactivity Insertion Accident
SAFDL	- Specified Acceptable Fuel Design Limit
SRP	- Standard Review Plan
SSE	- Safe-Shutdown Earthquake
TER	- Technical Evaluation Report

CONTENTS

1.0 INTRODUCTION

Framatome Cogema Fuels (FCF) has submitted to the NRC a topical report, entitled "Mark-B11 Fuel Assembly Design Topical Report" BAW-10229P (Reference 1), for review and approval. Presented in Reference 1 is the information required to support the licensing basis for the implementation of the Mark-B11 fuel assembly as reload fuel in Babcock and Wilcox pressurized water reactors (PWRs). This Technical Evaluation Report (TER) will address whether this new fuel design meets the NRC approved FCF fuel design criteria (Reference 2) and that the FCF analysis methodology used for this design applies to the Mark-B11 design up to the NRC approved rod average burnup level of 62 GWd/MTU (Reference 3).

It should be explained that Framatome Cogema Fuels was previously named the B&W Fuel Company (BWFC) a part of B&W Nuclear Technologies and prior to BWFC was named Babcock & Wilcox (B&W). Some of the references in this TER refer to these different company names depending on the date the reference was generated.

Pacific Northwest National Laboratory (PNNL) has acted as a consultant to the NRC in this review. As a result of the NRC staff's and their PNNL consultant's review of the topical report, a request for additional information (RAI) was sent by the NRC to FCF (Reference 5) requesting clarification of the design changes, lead test assembly data, the applicability of FCF evaluation methodology, and results of licensing analyses for the Mark-B11 design. FCF responded to those questions in Reference 6. FCF was further questioned for clarification of their responses in a January 26, 1999, conference call with NRC and PNNL. This conference call clarified their responses.

This review was based on those licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP) (Reference 7) and the FCF approved fuel design criteria (Reference 2). The objectives of this fuel system safety review, as described in Section 4.2 of the SRP, are to provide assurance that 1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), 2) fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) coolability is always maintained. A "not damaged" fuel system is defined as fuel rods that do not fail, fuel system dimensions that remain within operational tolerances, and functional capabilities that are not reduced below those assumed in the safety analysis. Objective 1, above, is consistent with General Design Criterion (GDC) 10 [10 Code of Federal Regulations (CFR) 50, Appendix A] (Reference 8), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 (Reference 9) for postulated accidents. "Coolable geometry," means in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat for design basis accidents. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident (LOCA) are given in 10 CFR 50, Section 50.46.

In order to assure that the above stated objectives are met and follow the format of Section 4.2 of the SRP, this review covers the following three major categories: 1) Fuel System Damage Mechanisms, which are most applicable to normal operation and AOOs; 2) Fuel Rod Failure Mechanisms, which apply to normal operation, AOOs, and postulated accidents; and 3) Fuel Coolability, which is applied to postulated accidents. Specific fuel damage or failure criteria are identified under each of these categories in Section 4.2 of the SRP. The FCF fuel design criteria or SAFDLs and the applicability of FCF analysis methodologies to the Mark-B11 design are discussed in this TER under each fuel damage or failure mechanism listed in the SRP.

The purpose of the design bases and/or criteria is to provide limiting values that prevent fuel damage or failure with respect to each mechanism. Reviewed in this TER is the applicability of the Mark-B11 design submitted in BAW-10229P to the FCF fuel design criteria and the applicability of FCF analysis methodologies to the Mark-B11 design are discussed. The FCF design criteria, along with certain definitions for fuel failure, constitute the SAFDLs required by GDC 10. The FCF analysis methods assure that the design limits and, thus, SAFDLs are met for a particular design application.

A description of a Mark-B11 fuel assembly is briefly discussed in the following section (Section 2.0). The fuel damage and failure mechanisms are addressed in Sections 3.0 and 4.0, respectively, while fuel coolability is addressed in Section 5.0.

2.0 FUEL SYSTEM DESIGN

The Mark-B11 fuel assembly consists of a 15x15 square array of fuel rods, control rod guide tubes, and a central instrumentation tube. The control rod guide tubes, central instrumentation tube, and eight spacer grids are mechanically fastened together with the top and bottom nozzles that make up the structural cage for the fuel rod assemblies. Fuel rods are supported at intervals along their length by the spacer grids with grid springs and dimples contained within the spacer grids to maintain rod-to-rod spacing. The spacer grid consists of an egg-crate arrangement of interlocking straps that contain springs and dimples that hold the fuel rods in place. The top nozzle is designed to allow for fuel assembly reconstitution, the same as for the Mark-B10 assembly. Attached to the top nozzle are holddown springs and spring clamps which keep the fuel assembly firmly seated on the lower core plate during normal plant operation.

The main differences between the Mark-B11 design and the Mark-B10 design is in the smaller diameter fuel rods, the use of flow mixing vanes on five of the six intermediate Zircaloy grids, and an improved grid restraint system on the central instrument tube. Due to the smaller diameter fuel rods the spacer grid cell size was reduced proportionately in the spacer grids in order to maintain the same spacer spring loads. All but the bottom intermediate spacer grids (five out of six) have the bent out vanes on the top of the grid interior strips. These vanes provide improved thermal performance by locally increasing the intensity of flow turbulence in the subchannel. Mixing vanes are not used on the lower intermediate grid since they are not needed in this cooler axial region of the assembly. A similar mixing vane grid is used in the Mark-B11 design for Westinghouse plants.

Due to the mixing vanes creating greater flow resistance in the uppermost intermediate grids there are greater loads placed on the grid restraint system. As a result the grid restraint system was redesigned to 1) increase the load-carrying capacity of the restraint system, and 2) to divide the loads between those from the lowest two intermediate spacer grids and those from the four uppermost intermediate spacer grids. The latter change reduces the loads on the uppermost sleeves that carry the increased loads due to the mixing vanes.

3.0 FUEL SYSTEM DAMAGE

The design criteria presented in this section should not be exceeded during normal operation including AOOs. The evaluation portion of each damage mechanism evaluates the analysis methods used by FCF to demonstrate that the design criteria are not exceeded during normal operation including AOOs for the reconstituted fuel assembly design.

3.1 STRESS

Bases/Criteria - In keeping with the GDC 10 SAFDLs, fuel damage criteria for cladding stress should ensure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The FCF design basis for fuel rod cladding stresses is that the fuel system will be functional and will not be damaged due to excessive stresses. The FCF criteria are based on guidelines established in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Reference 10). These criteria are consistent with the acceptance criteria established in Section 4.2 of the SRP and have been previously approved by NRC for Mark-B designs (Reference 2). These stress criteria are also acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The stress analyses for the Mark-B11 fuel assembly components and fuel rod cladding are based on standard engineering stress analysis methods including finite-element analysis and calculated in accordance with the ASME code, which includes both normal and shear stress effects. Pressure and temperature inputs to the stress analyses are chosen so that the operating conditions for all normal operation and AOOs are enveloped. The input cladding wall thicknesses are reduced to those minimum values allowed by fabrication specifications and further reduced by a conservative amount to allow for corrosion on the cladding inside and outside surfaces. These stress analysis methods have been approved for Mark B designs (Reference 2). PNNL concludes that the Mark-B stress analysis methods are acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod average).

FCF has performed bounding stress analyses using these methods that determined that the Mark-B11 design components, including the fuel rods, meet the approved FCF stress criteria. Therefore, PNNL further concludes that the Mark-B11 design is acceptable with respect to design stress analysis.

3.2 STRAIN

Bases/Criteria - The FCF design criterion for fuel rod cladding strain is that maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%. This criterion is intended to preclude excessive cladding deformation from normal operation and AOOs. This is the same criterion for cladding strain that is used in Section 4.2 of the SRP and has been previously approved by NRC (Reference 2). This strain criterion is also acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average).

The material property that could have a significant impact on the cladding strain limit at extended burnup levels is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burnup operation, to levels that would allow cladding failure without the 1% cladding strain criteria being exceeded under normal operation and AOOs. Recent out-of-reactor measured elastic and plastic cladding strain values from high burnup cladding from two PWR fuel vendors (References 11, 12 and 13) have shown a decrease in cladding ductilities when local burnups exceed 52 GWd/MTU and with increasing hydrogen (corrosion) levels. In addition, the majority of the high burnup data (tensile or burst test) shows that when hydrogen levels start to exceed 700 ppm the uniform strains begin to fall below 1%. As a result FCF has adopted a limit on maximum cladding corrosion that is consistent with maintaining cladding hydrogen levels below 700 ppm, and that has been approved by NRC (Reference 3). This is also found to be applicable to the Mark-B11 fuel design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The FCF strain analysis methods for Mark-B designs have been approved for application to Mark-B designs (Reference 2 and up to rod-average burnups of 62 GWd/MTU (Reference 3). FCF has performed bounding fuel rod cladding strain analyses using these methods that determined that the Mark-B11 design meets the above strain criterion within the design operating limits. PNNL concludes that FCF strain analysis methods are applicable to the Mark-B11 design and that the design is acceptable with respect to cladding strain up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

3.3 STRAIN FATIGUE

Bases/Criteria - The FCF design criterion for cladding strain fatigue is that the cumulative fatigue factor be less than 0.9 when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, which ever is the most conservative, is imposed as per the O'Donnell and Langer design curve (Reference 14) for fatigue usage. This criterion is consistent with that described in Section 4.2 of the SRP and has previously been approved (References 2 and 3). This strain fatigue criterion is also acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The FCF strain fatigue analysis methods for Mark-B designs have been approved for application to rod-average burnups of 62 GWd/MTU (References 2 and 3). FCF has performed bounding fuel rod cladding strain fatigue analyses using these methods that determined that the Mark-B11 design meets the above strain fatigue criterion within the design's operating limits. PNNL concludes that FCF strain fatigue analysis methods are applicable to the

Mark-B11 design and that the design is acceptable with respect to cladding strain fatigue up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

3.4 FRETTING WEAR

Bases/Criteria - Fretting wear is a concern for fuel, burnable poison rods, and guide tubes. Fretting, or wear, may occur on the fuel and/or burnable poison cladding surfaces in contact with the spacer grids if there is a gap between the grid spacer springs and the fuel rods or due to flow induced vibratory forces. The FCF design criterion for fretting wear is that the assembly design shall provide sufficient support to limit rod vibration and fretting wear. This criterion is consistent with Section 4.2 of the SRP and has previously been approved for Mark-B designs up to rod-average burnups of 62 GWd/MTU (References 2 and 3). This fretting wear criterion is also acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - FCF has performed extensive flow-induced vibration testing of the Mark-B11 fuel assembly to examine the vibrational response and to verify that no flow related vibrational phenomena existed that could result in fretting wear. The vibrational response of the Mark-B11 was compared to the vibrational response of the proven in-reactor performance of the Mark-B10 assembly. The comparisons were performed under a wide range of flow conditions that could be experienced in-reactor with both assembly types having comparable vibrational responses and very low amplitudes of vibration.

FCF has also performed a 1000 hour wear test of the Mark-B11 assembly at simulated full power operating conditions of temperature, pressure, flow and coolant chemistry. The grid springs of the spacer grids in this assembly were relaxed to simulate end-of-life conditions between the springs and fuel rods. The results of this test showed that the wear between the grid springs and fuel rods was less than those of previous Mark-B designs for the same test conditions. FCF has also pointed out that they have not seen any evidence of fretting wear in Mark-B11 lead test assemblies (LTAs) after one cycle of operation.

FCF was questioned (Reference 5) on the cross flow conditions of a mixed core with the Mark-B11 assemblies and whether these cross flows could result in sufficient forces to induce fuel rod vibration. FCF responded (Reference 6) that they had used the LYNXT model to investigate cross flow velocities in a mixed core and found that the maximum cross flow velocities were significantly less than those experienced at the core periphery for Mark-B cores with similar pressure drop characteristics. These results suggest that cross flow velocities between different Mark-B assemblies will not result in fretting wear.

Based on the above testing and analyses, PNNL concludes that the Mark-B11 design is acceptable with respect to fretting wear up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

3.5 OXIDATION AND CRUD BUILDUP

Bases/Criteria - Section 4.2 of the SRP identifies cladding oxidation and crud buildup as potential fuel system damage mechanisms. The SRP does not establish specific limits on cladding oxidation and crud buildup but does specify that their effects be accounted for in the thermal and mechanical analyses performed for the fuel. As noted in Section 3.2, the cladding ductility can be significantly decreased at higher burnup levels where oxide thickness and hydrogen levels can become relatively large because of accelerated corrosion at rod-average burnups above 50 to 55 GWd/MTU. As a result FCF has adopted a limit of 100 microns on maximum cladding corrosion that is consistent with maintaining cladding hydrogen levels below 700 ppm and has been previously approved (Reference 3). This maximum corrosion limit is based on a localized axial position on a fuel rod. PNNL concludes that this maximum corrosion limit is applicable to and acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - Section 4.2 of the SRP states that the effects of cladding crud and oxidation needs to be addressed in safety and design analyses, such as in the thermal and mechanical analysis. The amount of cladding oxidation is dependent on the cladding type, fuel rod powers, water chemistry control and primary inlet coolant temperatures, but the amount of oxidation and crud buildup increases with burnup and cannot be eliminated. Therefore, extended burnups result in a thicker oxide layer that provides an extra thermal barrier, cladding thinning and ductility decrease that can affect the mechanical performance. The degree of this effect is dependent on cladding type, reactor coolant temperatures, power history, and the level of success of a reactors' water chemistry program. The following is an evaluation of the FCF corrosion model.

FCF has adopted a new cladding corrosion model, COROSO2 (Reference 3), that is more conservative, i.e., predicts more corrosion, than the original OXIDEPC model in TACO3 and predicts the accelerated corrosion observed in high burnup rods much better than the OXIDEPC model. This model has been approved by NRC with the commitment by FCF to collect more maximum corrosion thickness data in the future (Reference 3). The Mark-B11 and the similarly designed Mark-BW LTAs will also provide corrosion data up to extended burnup levels (see Section 6.0 on Fuel Surveillance) to verify the applicability of the new corrosion model to the Mark-B11 design. The best estimate or slightly conservative prediction of the COROSO2 model is considered to be acceptable because of the conservatism in the FCF maximum corrosion limit. Based on FCFs commitment to collect corrosion data at extended burnup levels from their Mark-B and Mark-BW LTAs, PNNL concludes that the COROSO2 model is acceptable for application to the Mark-B11 design in predicting maximum corrosion levels up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average).

It is noted that FCF performs reload/cycle specific evaluations to verify that cladding corrosion is within their design limit. These cycle specific evaluations are not within the scope of this review.

3.6 ROD BOWING

Bases/Criteria - Fuel and burnable poison rod bowing are phenomena that alter the design-pitch dimensions between adjacent rods. Bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than place design limits on the amount of bowing that is permitted, the effects of bowing are included in the departure from nucleate boiling ratio (DNBR) analysis by a DNBR penalty when rod bow is greater than a predetermined amount. This approach is consistent with Section 4.2 of the SRP and has previously been approved for Mark-B designs up to a rod-average burnup of 62 GWd/MTU (References 2 and 3). This rod bowing criterion is also acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The FCF methodology for rod bowing analysis has been found to be very conservative for current Mark-B designs up to a rod-average burnup of 62 GWd/MTU (Reference 3). Rod bowing has been found to be dependent on the distance between grid spacers, the rod moment of inertia, material characteristics of the cladding, and flux distribution. The moment of inertia has changed a small amount with the change in cladding diameter but the effect on the rod bowing for the Mark-B11 assembly should be insignificant or a slight improvement. In addition, FCF intends to collect rod bow data from the Mark-B11 LTAs to confirm that the current FCF methodology remains conservative. Based on FCFs commitment to collect rod bow data from their Mark-B11 LTAs, PNNL concludes that FCF rod bow analysis methods are applicable to the Mark-B11 design up to the current Mark-B burnup operating limit of 62 GWd/MTU (rod-average).

3.7 AXIAL GROWTH

Bases/Criteria - The FCF design basis for axial growth is that adequate clearance be maintained between the fuel rod end-cap-shoulder and the top and bottom nozzles, i.e., shoulder gap clearance, to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly. Similarly, for assembly growth, FCF has a design basis that axial clearance between core plates and the bottom and top assembly nozzles should allow sufficient margin for fuel assembly irradiation growth during the assembly lifetime. These bases are consistent with Section 4.2 of the SRP and have previously been approved (References 2 and 3). These bases are also acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The FCF models used to predict shoulder gap clearance and assembly growth are based on gap clearance data and axial growth data from Mark-B and Mark-BW designs and FCF claims that they are applicable to those for the Mark-B11 design. FCF was questioned (Reference 5) on the applicability of this data to the Mark-B11 design and was requested to provide their one cycle shoulder gap clearance and growth data for comparison to those data from the earlier designs. They were also requested to provide the margin to shoulder gap closure and the margin for compressing the cruciform holddown springs to solid height up to a rod-average burnup of 62 GWd/MTU.

The FCF response (Reference 6) presented one cycle data from the Mark-B11 LTAs that indicated that the Mark-B11 shoulder gap and assembly growth data were within the scatter of the earlier Mark-B and Mark-BW data. FCF also provided the margins requested showing that both the margins for shoulder gap closure and solid compression of the holddown springs were relatively small up to a rod-average burnup of 62 GWd/MTU and 64 GWd/MTU, respectively. However, examination of the FCF analysis methods used for predicting shoulder gap clearances and assembly growth demonstrate that they are very conservative. For example, the FCF bounding curves used for both of these analyses are significantly greater than the 95/95 bounds of the data. Therefore, the actual margins to the design bases for axial growth are quite large. In addition, FCF intends to collect axial growth and shoulder gap clearance data from the Mark-B11 LTAs. PNNL concludes that these axial growth analysis methods are conservative. Therefore, PNNL further concludes that they are acceptable for application to the Mark-B11 design and that the design is acceptable with respect to axial growth up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

3.8 ROD INTERNAL PRESSURE

Bases/Criteria - Rod internal pressure is a driving force for, rather than a direct mechanism of, fuel system damage that could contribute to the loss of dimensional stability and cladding integrity. Section 4.2 of the SRP presents a rod pressure limit of maintaining rod pressures below system pressure that is sufficient to preclude fuel damage. The FCF design basis for the fuel rod internal pressure is that the fuel system will not be damaged due to excessive fuel rod internal pressure and FCF has established the "Fuel Rod Pressure Criterion" (Reference 15) to provide assurance that this design basis is met. These criteria are that the internal pressure of the FCF lead fuel rod in the reactor is limited to a value below which could cause 1) the diametral gap to increase due to outward cladding creep during steady-state operation, and 2) extensive DNB propagation to occur. This FCF design basis and the associated criteria have been found acceptable by the NRC (Reference 15) up to the current Mark-B burnup limits established in Reference 3. PNNL concludes these are also acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - FCF utilizes the approved TACO3 fuel performance code (Reference 16) for predicting end-of-life (EOL) fuel rod pressures and the methodology described in Reference 15 to verify that they do not exceed the FCF "Fuel Rod Pressure Criterion" during normal operation and AOOs. The TACO3 fuel performance code is generic enough to be applicable to all FCF PWR fuel designs, and therefore is acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average). The issue of DNB propagation (Fuel Rod Pressure Criterion 2 above) will be discussed in Section 4.3. The FCF rod pressure analyses are performed on a reload/cycle specific basis.

3.9 ASSEMBLY LIFTOFF

Bases/Criteria - Section 4.2 of the SRP calls for the fuel assembly holddown capability (wet weight and spring forces) to exceed worst case hydraulic loads for normal operation and AOOs. The FCF design criterion for assembly liftoff is that the holddown spring system shall be capable of maintaining fuel assembly contact with the lower support plate during normal operation and AOOs. This is consistent with the SRP guidelines and has previously been approved (References 2 and 3). This criterion is also acceptable for application to the Mark-B11 design up to the current Mark B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The fuel assembly liftoff forces are a function of primary coolant flow, holddown spring forces, assembly dimensional changes and friction pressure drop across the length of the assembly with the spacer grids a major contributor to the pressure drops. FCF has performed several hydraulic tests in a full scale flow facility to measure the pressure drop characteristics of the Mark-B11 fuel assembly which were used to calculate the form loss coefficients.

FCF has performed several analyses of hydraulic lift forces using the form loss coefficients for a Mark-B11 assembly in both a full core and mixed core environment that demonstrates that the Mark-B11 assembly has lower lift forces than a Mark-B10 assembly for both core environments. This demonstrates that the Mark-B11 lift loads are bounded by the Mark-B10 values. PNNL concludes that FCF has performed adequate testing and analyses to verify the lift forces for the Mark-B11 design meet the FCF design criterion and, therefore, this issue has been adequately addressed.

4.0 FUEL ROD FAILURE

In the following paragraphs, fuel rod failure thresholds and analysis methods for the failure mechanisms listed in the SRP will be reviewed. When the failure thresholds are applied to normal operation including AOOs, they are used as limits (and hence SAFDLs) since fuel failure under those conditions should not occur according to the traditional conservative interpretation of GDC 10. When these thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose assessments required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus established by GDC 10 and Part 100 and only the threshold values and the analysis methods used to assure that they are met are reviewed below.

4.1 HYDRIDING

Bases/Criteria - Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities in the fuel during fabrication; this is generally an early-in-life failure mechanism. FCF has not discussed their criteria for internal hydriding in the subject topical report; however, a limit on hydrogen level for FCF pellets is discussed in Reference 17. The hydrogen level of FCF fuel pellets is controlled by drying the pellets in the cladding and taking a statistical sample to ensure that the hydrogen level is below a specified level. Previous FCF design reviews, e.g., Reference 17, have shown that this level is below the value recommended in the SRP. Consequently, PNNL concludes that the FCF limit on hydrogen in their fuel pellets is acceptable for the Mark-B11 design.

External hydriding of the cladding due to waterside corrosion is the other source and is discussed in Section 3.5 of this TER. As noted in Section 3.5, the level of external hydriding is controlled by FCF by a proprietary limit on corrosion thickness. PNNL concludes that this corrosion limit is acceptable for limiting the level of external hydriding in the cladding for the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - Internal hydriding is controlled by FCF by taking statistical samples following pellet fabrication prior to loading the pellets in the fuel rods and confirming that hydrogen is below a specified level. Therefore, no analyses are necessary other than to confirm that the statistical pellet sampling is below the specified level for Mark-B11 designs.

External hydriding is controlled by the FCF limit on corrosion thickness discussed in Section 3.5 of this TER.

PNNL concludes that FCF has addressed the issue of hydriding in Mark-B11 designs up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

4.2 CLADDING COLLAPSE

Bases/Criteria - If axial gaps in the fuel pellet column were to occur due to fuel densification, the potential would exist for the cladding to collapse into a gap. Because of the large local strains that would result from collapse, the cladding is then assumed to fail. The FCF design criterion is that cladding collapse is precluded during the fuel rod design lifetime. This design basis is the same as that in Section 4.2 of the SRP and has previously been approved (References 2 and 3). This criterion is also acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The FCF analytical models for evaluating cladding creep collapse are the CROV and TACO3 computer codes that have been reviewed and approved by NRC (References 18 and 16). FCF has provided the results of their bounding creep collapse analysis that demonstrates that collapse will not occur for the Mark-B11 design up to a rod-average burnup of 70 GWd/MTU using a conservatively high average power history. PNNL concludes that these codes and methods are conservative for evaluating cladding creep collapse in FCF PWR designs and, therefore, are acceptable for application to the Mark-B11 design. Based on the FCF analyses, PNNL further concludes that the Mark-B11 design is acceptable with respect to cladding collapse up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

4.3 OVERHEATING OF CLADDING

Bases/Criteria - The FCF design limit for the prevention of fuel failures due to cladding overheating is that there will be at least a 95% probability at a 95% confidence level that departure from nucleate boiling (DNB) will not occur on a fuel rod having the minimum DNBR during normal operation and AOOs. This design limit is consistent with the thermal margin criterion of Section 4.2 of the SRP and has previously been approved for FCF designs (References 2 and 3). This design limit is also acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - As stated in the SRP, Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion to limit DNB or boiling transition in the core is satisfied. FCF has submitted a new CHF correlation for the Mark-B11 design. FCF utilizes NRC-approved critical heat flux (CHF) correlations for evaluating thermal margins and these analyses are performed on a reload/cycle specific basis.

As noted in Section 3.8, one of the design criteria for rod pressures is that the limit on rod pressures prevent extensive DNB propagation to occur. The FCF methodology for evaluating DNB propagation is described in Reference 15 and has been approved by NRC. PNNL concludes that this FCF analysis methodology for preventing DNB propagation due to rod over-pressures is acceptable for application to the Mark-B11 design.

4.4 OVERHEATING OF FUEL PELLETS

Bases/Criteria - As a second method of avoiding cladding failure due to overheating, FCF precludes centerline pellet melting during normal operation and AOOs. This design criterion is the same as that given in the SRP and has previously been approved for FCF designs up to current operating limits (References 2 and 3). This criterion for fuel melting is also acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - FCF utilizes the approved TACO-3 fuel performance code to determine the maximum linear heat generation rate (LHGR) at which a given fuel design will not achieve fuel melting at a 95% probability at a 95% confidence level. This FCF analysis methodology has been found to be acceptable to Mark-B designs up (Reference 2) to a rod-average burnup of 62 GWd/MTU (Reference 3). PNNL also finds them acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

FCF has also performed a fuel melting analysis for the Mark-B11 fuel design that demonstrates that the Mark-B11 design is acceptable within the design's operating limits. PNNL concludes that the Mark-B11 design is acceptable in relation to fuel melting up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

4.5 PELLET/CLADDING INTERACTION

Bases/Criteria - As indicated in Section 4.2 of the SRP, there are no generally applicable criteria for pellet cladding interaction (PCI) failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: 1) less than 1% transient induced cladding strain, and 2) no centerline fuel melting. Both of these limits have been adopted by FCF for use in evaluating their fuel designs and have been approved by the NRC. These two criteria have been satisfactorily addressed in Sections 3.2 and 4.4 of this TER and will not be discussed further in this section.

Evaluation - As noted earlier, FCF utilizes the TACO-3 (Reference 16) code to show that their fuel meets both the cladding strain and fuel melting criteria. This code is acceptable per the recommendations in Sections 3.2 and 4.4.

4.6 CLADDING RUPTURE

Bases/Criteria - There are no specific design limits associated with cladding rupture other than the 10 CFR 50, Appendix K (Reference 19) requirements that the incidence of rupture not be underestimated. FCF uses a rupture temperature correlation consistent with NUREG-0630 guidance (Reference 20). PNNL concludes that FCF has adequately addressed cladding rupture for the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - FCF has adopted the cladding deformation and rupture models from NUREG-0630 guidance (Reference 20) which has been approved by the NRC for ECCS evaluation. PNNL concludes that FCF has adequately addressed the issue of cladding rupture for the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

4.7 FUEL ROD MECHANICAL FRACTURING

Bases/Criteria - The term "mechanical fracture" refers to a fuel rod defect that is caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. The design limits proposed by FCF to prevent fracturing is that the stresses due to postulated accidents in combination with the normal steady-state fuel rod stresses should not exceed the stress limits established in the approved methodology (Reference 2) for Mark-B fuel assembly designs. These design limits for fuel rod mechanical fracturing are acceptable for application to the Mark-B11 fuel design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

Evaluation - The mechanical fracturing analysis is done as a part of the seismic-and-LOCA loading analysis. A discussion of the seismic-and-LOCA loading analysis is given in Section 5.4 of this TER.

5.0 FUEL COOLABILITY

For postulated accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDCs (e.g., GDC 27 and 35). In the following paragraphs, limits and methods used to assure that coolability is maintained are discussed for the severe damage mechanisms listed in the SRP.

5.1 FRAGMENTATION OF EMBRITTLED CLADDING

Bases/Criteria - The most severe occurrence of cladding oxidation and possible fragmentation during a postulated accident is the result of a LOCA. FCF has not discussed cladding embrittlement as a result of a LOCA in the subject topical report but this has been previously presented by FCF in References 2 and 3 that have been approved by NRC. In order to reduce the effects of cladding oxidation during LOCA, FCF uses a limiting criteria of 2200°F on peak cladding temperature (PCT) and a limit of 17% on maximum cladding oxidation as prescribed in 10 CFR 50.46 and consistent with the SRP criteria. PNNL concludes that these criteria are also applicable to the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU.

Evaluation - FCF has evaluated the impact of the Mark-B11 design changes on LOCA utilizing approved LOCA analysis methods. This analysis concluded that the Mark B-11 design meets the requirements of 10 CFR 50.46, and FCF will confirm this on a plant-specific basis.

5.2 VIOLENT EXPULSION OF FUEL

Bases/Criteria - In a severe reactivity insertion accident (RIA), such as a control rod ejection accident, large and rapid deposition of energy in the fuel could result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal might be sufficient to destroy the fuel cladding and rod bundle geometry and provide significant pressure pulses in the primary system. To limit the effects of an RIA event, Regulatory Guide 1.77 (Reference 21) recommends that the radially-averaged energy deposition at the hottest axial location be restricted to less than 280 cal/g and the onset of DNB is assumed to be the failure limit. It is noted that the NRC staff are currently reviewing the 280 cal/gm limit and the limit for fuel failure may be decreased to a lower limit at high burnup levels. Recent RIA testing has indicated that fuel expulsion and fuel failure may occur before the 280 cal/gm limit and the onset of DNB, respectively (References 22 and 23). However, further testing and evaluation is needed to establish limits. The fuel expulsion and failure limits for an RIA may decrease in the future but the current limits remain valid at this time.

The FCF design criterion for this event is identical to that in Regulatory Guide 1.77, such that the peak fuel enthalpy for the hottest axial fuel rod location shall not exceed 280 cal/gm. Therefore, PNNL concludes that FCF design limits for fuel dispersal are acceptable for application to the Mark-B11 design up to the current Mark-B operating burnup limit of

62 GWd/MTU.

Evaluation - FCF verifies that this acceptance criterion is met for each fuel cycle through design and cycle specific analyses and by limiting the ejected rod worth. FCF uses NRC-approved methods to perform these analyses and the methods remain valid for the Mark-B11 design. PNNL concludes that the analysis methodology remains acceptable for application to the Mark-B11 fuel design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

5.3 CLADDING BALLOONING

Bases/Criteria - Fuel cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. There are no specific design limits associated with cladding ballooning other than the 10 CFR 50 Appendix K requirement that the degree of swelling not be underestimated.

Evaluation - The cladding ballooning model and flow blockage model are directly coupled to the cladding rupture temperature model for the LOCA-emergency core cooling system (ECCS) analysis that is plant specific. FCF has adopted the cladding rupture and ballooning models from NUREG-0630 (Reference 20) as recommended by Section 4.2 of the SRP and these models have been previously approved by the NRC. Therefore, PNNL concludes that FCF has adequately addressed the issue of cladding ballooning and that these models remain acceptable for application to Mark-B11 designs up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

5.4 FUEL ASSEMBLY STRUCTURAL DAMAGE FROM EXTERNAL FORCES

Bases/Criteria - Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. Appendix A to SRP Section 4.2 states that the fuel system coolable geometry shall be maintained and damage should not be so severe as to prevent control rod insertion during seismic and LOCA events. The FCF design basis is that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case design accident and that no interference between control rods and thimble tubes will occur during a safe shutdown earthquake. This is consistent with the SRP and is therefore acceptable for application to the Mark-B11 fuel design up to the current Mark-B operating limits.

Evaluation - FCF has performed impact tests on the Mark-B11 spacer grids to characterize the plastic deformation and elastic limits of the spacer grids. These tests show that the Mark-B11 spacer grids are slightly stronger than the previous Mark-B Zircaloy grids. FCF has also performed dynamic pluck, axial stiffness and lateral stiffness tests on the Mark-B11 assembly that determined that the natural frequency, and axial and lateral stiffness values were close to those of previous Mark-B assemblies with Zircaloy grids.

FCF has performed a seismic-LOCA analysis using approved analysis methods to determine the Mark-B11 fuel assembly structural response to bounding seismic-LOCA loadings. These analyses demonstrate that the grid spacer loadings are well within their elastic limits and, therefore, the assembly retains a coolable geometry. Consequently, PNNL concludes that FCF has satisfactorily addressed the issue of seismic-LOCA loads for the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

6.0 FUEL SURVEILLANCE

FCF was questioned about what future fuel surveillance would be performed to verify satisfactory performance of the Mark-B11. FCF responded that their lead test assembly (LTA) program consisted of four Mark-B11 fuel assemblies being irradiated in Oconee-2. Three of the four assemblies will be irradiated for two cycles (assembly average burnup of 25 GWd/MTU) and one assembly for three cycles (assembly average burnup of 39 GWd/MTU). The LTAs will be placed in positions in the core periphery (where previous fretting had been observed) during the second cycle in order to demonstrate that the new spacer grids are not susceptible to fretting wear. Each Mark-B11 LTA will be subjected to the following inspections; visual, fuel assembly length and bow, guide tube distortion, spacer grid width, and fuel rod shoulder gap clearances. The oxide thickness of the fuel rods, guide tubes, and spacer grids will also be measured.

PNNL verbally questioned FCF about the lack of high burnup Mark-B11 data, i.e., above an assembly average burnup of 39 GWd/MTU, particularly in regards to cladding corrosion because this is one of the burnup limiting parameters for FCF fuel designs. FCF responded that the mixing vane grid design in Mark-B11 is essentially the same as used in the Mark-BW designs from which they have higher burnup data and also from European fuel designs with mixing vane grids. FCF has cladding oxidation data from the Mark-BW design up to rod-average burnups of 54 GWd/MTU that demonstrate that their COROSO2 corrosion model adequately predicts cladding corrosion, and therefore, it is expected that it will also adequately predict cladding corrosion for the Mark-B11 design up to the current Mark-B operating burnup limit of 62 GWd/MTU (rod-average).

PNNL concludes that FCF has adequately addressed the issue of fuel surveillance.

7.0 CONCLUSIONS

PNNL has reviewed the FCF thermal-mechanical design criteria and analyses for the Mark-B11 fuel design presented in Reference 1 in accordance with Section 4.2 of the SRP. PNNL concludes that the Mark-B11 design as described in Reference 1 is acceptable for reload licensing applications up to a rod-average burnup of 62 GWd/MTU.

. As noted in Section 4.3 of this TER the critical heat flux correlation for the Mark-B11 design is still under review and needs to be approved before the design can be used in reload applications. For those licensees that apply this reload methodology, the following plant-specific analyses or evaluations are required: 1) cladding oxidation (Section 3.5); 2) rod internal pressures (Section 3.8); 3) overheating of cladding (Section 4.3); and 4) ECCS related analyses (Sections 5.1, 5.2, and 5.3).

8.0 REFERENCES

1. Framatome Cogema Fuels. September 1997. Mark-B11 Fuel Assembly Design Topical Report. BAW-10229P, Framatome Cogema Fuels, Lynchburg, Virginia, transmitted by letter, J. H. Taylor (FCF) to U.S. NRC Document Control Desk, "Submittal of Topical Report BAW-10229P, Mark-B11 Fuel Assembly Design Report," dated September 30, 1997, JHT/97-35.

2. B&W Fuel Company. February 1991. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses. BAW-10179P, B&W Fuel Company, Lynchburg, Virginia.

3. B&W Fuel Company. November 1992. Extended Burnup Evaluation. BAW-10186P, B&W Fuel Company, Lynchburg, Virginia.

4. Framatome Cogema Fuels. September 1996. The BWU Critical Heat Flux Correlations BAW-10199P, Addendum 1, Framatome Cogema Fuels, Lynchburg, Virginia.

5. Letter, J. L. Birmingham (NRC) to C. F. McPhatter (FCF), "Request for Additional Information for Topical Report BAW-10229P, Mark-B11 Fuel Assembly Design Report," dated September 8, 1998.

6. Letter, T. A. Coleman (FCF) to U.S. NRC Document Control Desk, dated November 13, 1998, GR855.doc.

7. U.S. Nuclear Regulatory Commission. July 1981. "Section 4.2, Fuel System Design." In Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition. NUREG-0800, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

8. United States Federal Register. "Appendix A, General Design Criteria for Nuclear Power Plants." 10 Code of Federal Regulations (CFR), Part 50. U.S. Printing Office, Washington, D.C.

9. United States Federal Register. "Reactor Site Criteria." 10 Code of Federal Regulations (CFR), Part 100. U.S. Printing Office, Washington, D.C.

10. American Society of Mechanical Engineers. 1983 Edition. "Section III, Nuclear Power Plant Components." ASME Code. American Society of Mechanical Engineers, New York.

11. Smith, Jr. G. P., R. C. Pirek and M. Griffiths. July 1994. Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1. EPRI TR-103302-V2, Final Report, Electric Power Research Institute, Palo Alto, California.

12. Newman, L. W. et al. 1986. The Hot Cell Examination of Oconee Fuel Rods After Five Cycles of Irradiation. DOE/ET/34212-50 (BAW-1874), Babcock & Wilcox, Lynchburg, Virginia.

13. Garde, A. M. 1989. "Effects of Irradiation and Hydriding on the Mechanical Properties of Zircaloy-4 at High Fluence." In Zirconium in the Nuclear Industry: Eighth International Symposium, ASTM STP 1023, pp. 548-569, Eds. L.F.P. VanSwam and C. M. Eucken. American Society for Testing and Materials, Philadelphia, Pennsylvania.

14. O'Donnell, W. J., and B. F. Langer. 1964. "Fatigue Design Basis for Zircaloy Components." In Nuc. Sci. Eng. 20:1.

15. D. A. Wesley, D. A. Farnsworth and G.A. Meyer, July 1995. Fuel Rod Gas Pressure Criterion (FRGPC), BAW-10183P-A, B&W Fuel Company, Lynchburg, Virginia.

16. Wesley, D. A., and K. J. Firth. October 1989. TACO-3 Fuel Pin Thermal Analysis Code. BAW-10162P-A, Babcock & Wilcox, Lynchburg, Virginia.

17. Babcock and Wilcox. April 1986. Extended Burnup Evaluation. BAW-10153P-A, Babcock and Wilcox, Lynchburg, Virginia.

18. Miles, T., et. al., August 1995. Program to Determine In-Reactor Performance of B&W Fuel Cladding Creep Collapse, BAW-10084P-A Rev. 3, B&W Fuel Company, Lynchburg, Virginia.

19. United States Federal Register. "Appendix K, ECCS Evaluation Models." 10 Code of Federal Regulations (CFR), Part 50. U.S. Printing Office, Washington, D.C.

20. Powers, D. A., and R. O. Meyer. 1980. Cladding Swelling and Rupture Models for LOCA Analysis. NUREG-0630, U.S. Nuclear Regulatory Commission, Washington, D.C.

21. U.S. Nuclear Regulatory Commission. 1974. Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, Regulatory Guide 1.77, U.S. Nuclear Regulatory Commission, Washington D.C.

22. Schmitz, F., et. al., March 1996. "New Results from Pulse Test in the CABRI Reactor," Proceedings of the 23rd Water Reactor Safety Information Meeting October 23-25,1995.

23. Fuketa, T., et. al., March 1996. "New Results from the NSRR Experiments with High Burnup Fuel," Proceedings of the 23rd Water Reactor Safety Meeting, October 23-25, 1995.