FRAMATOME COGEMA FUELS

April 28, 2000
GR00-044.doc

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Submittal of Topical Report BAW-10229P, "Mark-B11 Fuel Assembly Design Report," April 2000.

Gentlemen:

Enclosed are fifteen (15) copies of Topical Report BAW-10229P-A and twelve (12) copies of Topical Report BAW-10229-A. These reports will serve as the accepted versions, proprietary and non-proprietary of BAW-10229P which was recently reviewed and found to be acceptable by the NRC staff. BAW-10229P provides the licensing bases for the Mark-B11 fuel assembly design.

Copies of the NRC acceptance letter and accompanying SER are included between the title page and the table of contents of the report. Copies of responses to the NRC request for additional information on BAW-10229P are included as Appendix A of the report.

In accordance with 10 CFR 2.790, FCF requests that BAW-10227P-A be considered proprietary and withheld from public disclosure. An affidavit supporting this request is attached.

Very truly yours,

T A Coleman

T. A. Coleman, Vice President
Government Relations

cc: J. S. Wermiel, NRC
S. L. Wu, NRC
S. N. Bailey, NRC
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BAW-10229-A
April 2000

MARK-B11 FUEL ASSEMBLY DESIGN TOPICAL REPORT

FRAMATOME COGEMA FUELS

BAW-10229-A
April 2000

MARK-B11 FUEL ASSEMBLY DESIGN TOPICAL REPORT

Framatome Cogema Fuels
P. O. Box 10935
3315 Old Forest Road
Lynchburg, Va. 24506-0935

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 26, 1999

Mr. T. A. Coleman, Vice President
Government Relations
Framatome Cogema Fuels
3315 Old Forest Road
P. O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: SAFETY EVALUATION FOR TOPICAL REPORT BAW-10229P, "MARK-B11 FUEL ASSEMBLY DESIGN TOPICAL REPORT," SEPTEMBER 1997 (TAC NO. M99904)

Dear Mr. Coleman:

The staff has reviewed the subject report submitted by Framatome Cogema Fuels (FCF) by letter of September 30, 1997, and additional information submitted by letter dated November 13, 1998, that was in response to our request for additional information. On the basis of our review, the staff has found the subject report to be acceptable for referencing in license applications to the extent specified, and under the limitations stated, in the enclosed safety evaluation (SE) and the technical evaluation report attached to the SE.

The staff will not repeat its review of the matters described in FCF Topical Report BAW-10229P and found acceptable when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in BAW-10229P.

In accordance with procedures established in NUREG-0390, the NRC requests that FCF publish accepted versions of the report including the safety evaluation, in proprietary and non-proprietary forms, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter, and the enclosed evaluation between the title page, and the abstract, and an -A (designating accepted) following the report identification symbol. The accepted versions shall also incorporate all communications between FCF and the staff during this review.

Should our acceptance criteria or regulations change so that our conclusions as to the acceptability of the report are no longer valid, applicants referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued applicability of the topical report without revision of their respective documentation.

This concludes NRC review activity for this report (TAC M99904). If you have any questions regarding this matter please contact me at (301) 415-1321, or by email at snb@nrc.gov.

Sincerely,

Stewart N. Bailey, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 693

Enclosure: Safety Evaluation

cc w/encl:
Mr. F. McPhatter, Manager
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT BAW-10229P

"MARK-B11 FUEL ASSEMBLY DESIGN TOPICAL REPORT"

1.0 INTRODUCTION

By letter dated September 30, 1997, Framatome Cogema Fuels (FCF) submitted Topical Report BAW-10229P, "Mark-B11 Fuel Assembly Design Topical Report," for NRC review.

BAW-10229P describes a new fuel assembly mechanical design, Mark-B11, for fuel reload licensing applications in pressurized-water reactors (PWRs). The Mark-B11 fuel design is very similar to the previously approved Mark-B fuel designs. The Mark-B11 fuel consists of a 15x15 square array of fuel rods, control rod guide tubes, and a central instrumentation tube. The main differences between the Mark-B11 fuel design and the earlier Mark-B fuel designs is the Mark-B11's use of smaller diameter fuel rods, flow mixing vanes on five of the six intermediate zircaloy grids, and an improved grid restraint system on the central instrumentation tube. The Mark-B11 fuel design also intends to improve its thermal-hydraulic performance.

The NRC staff was supported in this review by its consultant, Pacific Northwest National Laboratory (PNNL). PNNL's technical evaluation report (TER) is attached.

2.0 EVALUATION

The staff has reviewed the attached TER, and has determined that the TER describes the technical basis for approving BAW-10229P with the exception of TER Section 5.2, Violent Expulsion of Fuel. With regard to Section 5.2, the staff believes that additional clarification is necessary with respect to the acceptance criteria in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," and Standard Review Plan Section 4.2, "Fuel System Design," for the rod ejection accidents. These acceptance criteria are considered nonconservative in light of some test data from foreign test reactors on reactivity-initiated accidents. However, the staff considers the fuel to be acceptable to a rod-average burnup level of 62,000 Mwd/MTU burnup because the probability of these accidents is low and generic plant transient calculations indicate that energy inputs during these transients are low and will remain below the relevant test data failure levels. This position is consistent with the Agency Program Plan for High-Burnup Fuel and the memorandum from J. Callan to the Commissioners dated July 15, 1997.

The following plant-specific analyses will be required for those licensees applying the Mark-B11 fuel in reload fuel designs: (1) cladding oxidation (TER Section 3.5), (2) rod internal pressures (TER Section 3.8), (3) overheating of cladding (TER Section 4.3), and (4) ECCS related analyses (TER Sections 5.1, 5.2, and 5.3). In a letter dated July 27, 1999, from T. A. Coleman (FCF) to U. S. NRC, FCF confirmed that the above mentioned four items will be performed on a plant-specific basis for each reload application.

With the above clarification and plant-specific analyses requirements, the staff agrees with PNNL's conclusion that the Mark-B11 fuel assembly mechanical design described in BAW-10229P is acceptable for fuel reload licensing applications in PWRs up to a rod-average burnup of 62,000 MWd/MTU. Based on our review, the staff adopts the findings in the attached TER.

3.0 CONCLUSION

The staff has reviewed the FCF's Mark-B11 fuel assembly mechanical design described in BAW-10229P, and finds that the Mark-B11 fuel design is adequate and thus acceptable for fuel reload licensing applications up to 62,000 MWd/MTU rod average burnup in PWRs. Plant-specific analyses will be required for those licensees using the Mark-B11 fuel in reload fuel designs: (1) cladding oxidation, (2) rod internal pressures, (3) overheating of cladding, and (4) ECCS related analyses as described in the above Section 2.0 of this safety evaluation.

Attachment: Technical Evaluation Report

Principle Contributor: S. L. Wu

Date: October 26, 1999

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-AC06-76RLO 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).

5.3

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).

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1. INTRODUCTION

Following four years of thorough design development and testing, the Mark-B11 fuel assembly is the most recent addition to FCF's Mark-B fuel product line, utilized in Babcock & Wilcox (B&W) 177 fuel assembly-designed reactors. The Mark-B11 fuel design features a smaller-diameter fuel rod to reduce enriched uranium requirements for both transition and equilibrium cycles and mixing vane grids that provide superior thermal margins.

Four Mark-B11 lead assemblies have operated successfully since installation into cycle sixteen of Duke Power Oconee Nuclear Unit 2 reactor in April 1996. Subsequent batch implementation of the Mark-B11 fuel assembly design is planned for all three Duke Power Oconee Nuclear Units beginning with cycle nineteen of Oconee Nuclear Unit 3 in 1999.

The Mark-B11 fuel assembly is designed to achieve a peak fuel rod burnup of 62,000 MWd/mtU, which is consistent with the burnup limits approved in BAW-10186P-A, "Extended Burnup Evaluation" [1].

This topical report contains the licensing bases for the Mark-B11 fuel assembly which provide justification for batch implementation. This report is divided into eight major sections, each addressing a significant aspect of the Mark-B11 fuel assembly, focusing on the primary new features, which include the reduced fuel rod diameter, flow mixing intermediate grids, and improved grid restraint system. Section 3 describes the Mark-B11 design, highlighting the standard and new distinguishing features. Section 4 presents the scope and results of the fuel assembly and component design verification testing. Sections 5 and 6 provide the fuel assembly and fuel rod mechanical evaluations respectively, which address the key structural issues as affected by the primary Mark-B11 design features. The evaluation of the thermal-hydraulic performance of the Mark-B11 assembly is presented in section 7, which addresses the mixing grid and rod diameter effects. Sections 8 and 9 provide the nuclear design and ECCS evaluations, respectively. Section 10 is an overall assessment of the impact of the Mark-B11 fuel assemblies on plant operations.

2. SUMMARY

The Mark-B11 fuel assembly is a natural progression of the Mark-BZ fuel design which offers improvements in departure from nucleate boiling (DNB) margins and fuel cycle economy while possessing many proven features of earlier Mark-BZ fuel assembly designs. Proven features of the Mark-BZ fuel design utilized for the Mark-B11 design include keyable spacer grids, floating grid restraint system, flow-optimized control rod guide tube assembly, quick disconnect upper end fitting assembly, anti-straddle lower end fitting assembly, Zircaloy intermediate grids, cruciform holddown spring assembly, and debris resistant fuel rod lower end plug.

The specific Mark-B11 design features that enhance the design's nuclear, thermal-hydraulic and mechanical performance include the following:

1. Reduced diameter fuel rod,
2. Flow mixing vanes on five of the six intermediate spacer grids, and
3. Improved grid restraint system.

Improved thermal mixing with the mixing vane grids increases DNB margins, which provides for more aggressive fuel cycle designs. Increased uranium utilization is also gained through the use of the reduced fuel pin diameter, providing for improved fuel cycle economy. An improved grid restraint system provides additional structural strength to accommodate the increased hydraulic loads attributed to the flow mixing grids.

The Mark-B11 design verification program addressed key factors associated with the incorporation of the three primary features of the Mark-B11 assembly. The results from the prototype testing and analyses in the mechanical, thermal-hydraulic, core physics, and ECCS areas verify that the Mark-B11 fuel assembly is a safe and reliable design. The successful operation to date of the Mark-B11 lead test assemblies (LTAs) further supports the results of the design verification program. In addition, the extensive operating experience of the Mark-BZ and the Mark-BW (17x17 design for Westinghouse-designed reactors) designs provides a performance data base for many of the critical design features which are common to the Mark-B11 fuel assembly and all FCF fuel designs. These key features, which include the floating intermediate spacer grid and seated fuel rod design concepts, serve to provide well predicted and consistent irradiation performance and models and further enhance the Mark-B11 design bases.

Based on the results of extensive testing, analysis, and reactor performance, the Mark-B11 is acceptable for batch implementation in B&W designed Pressurized Water Reactors (PWRs).

3. MARK-B11 DESIGN DESCRIPTION

3.1 Fuel Assembly Design Description

The Mark-B11 fuel assembly comprises a 15x15 rod array specifically developed for use in B&W 177 fuel assembly designed nuclear reactors. The fuel assembly maintains the same interface compatibility and many of the reactor proven features of the resident Mark-BZ fuel. Figures 3.1 and 3.2 highlight the key design features of the Mark-B11 fuel assembly and fuel rod respectively, with those unique to the Mark-B11 design designated in bold type.

3.1.1 Standard Design Features

3.1.1.1 Fuel Assembly

The Mark-B11 (as is the Mark-BZ) is a conventional 15x15 fuel assembly designed specifically for Babcock & Wilcox-designed 177 fuel assembly pressurized water reactors (PWR). Within its 15x15 lattice arrangement are 16 low-tin Zircaloy-4 control rod guide tubes that attach to stainless steel upper and lower end fittings. The guide tubes contain side holes designed specifically to control guide tube bypass flow while providing adequate guide tube flow for control component cooling and guidance for control rod insertion. A full length low-tin Zircaloy-4 instrument tube occupies the center lattice position, which provides guidance for in core instrumentation and support for the grid restraint system.

The Mark-B11 fuel assembly utilizes eight spacer grids, which with the guide tubes, instrument tube, and end fittings, provide the structural cage for the Zircaloy clad fuel rod assemblies. The upper and lower end grid strips are made from Inconel 718. The six intermediate grids are constructed from fully annealed, low-tin Zircaloy-4. The remaining 208 lattice positions contain low-tin cold-worked stress-relieved Zircaloy-4 clad fuel rods that rest on the lower end fitting grillage and are laterally supported by the upper and lower end spacer grids and six intermediate spacer grids.

Just as with the Mark-BZ and all FCF designs, the Mark-B11 spacer grid design is keyable and utilizes hard/soft stops in the cells to support the fuel rod. The spacer grid consists of thin strips welded together in "egg crate" style forming an array of square cells. In each cell, protrusions or "stops" are formed into the cell walls. These cells are arranged in sets - hard stops on upper and lower edges to position the fuel rod, and a soft stop at mid-height of the opposite side to clamp the rod in place. A key holds the grid cells open during manufacturing so that the fuel rods can be slipped into the assembly, rather than being forced through the grids. The keying process prevents scratching or other damage to the fuel rod cladding. Once all the rods are in place, the keys are removed. This procedure also minimizes residual stresses in the rods as a result of manufacturing and thus serves to mitigate rod bow during operation. Mark-B11 end and intermediate grids maintain the same periphery lead-in features as used in the Mark-BZ design to ensure good fuel assembly-handling performance.

As with the Mark-BZ design, the Mark-B11 spacer grids are not mechanically attached to the control rod guide tubes. Thus, the grids are free to axially accommodate any differential growth between the fuel rods and guide tubes, i.e. free to "float". The spacer sleeves around the instrument tube are designed to control the vertical location of the intermediate grids. The vertical location of the spacer grids remains unchanged from previous Mark-BZ designs. This arrangement substantially reduces the axial forces on the guide tubes and fuel rods, and the resultant forces on the spacer grids. This feature is especially important during the early-in-life assembly operation when the fuel rod grip forces are relatively high. This feature coupled with the seated fuel rods serve to reduce guide tube distortion. Local distortion attributed to grid-to-guide tube fixity is minimized by the floating grids. Guide tube axial loads are reduced with the weight of the fuel rods passing directly to the lower end fitting thereby mitigating guide tube distortion.

Features on the guide tube assemblies constrain axial motion of the end grids. The bottom end grid is restrained through guide tube lower end plugs fixed to the lower end fitting. Upper end grid motion is restrained by spacer sleeves located on the guide tubes between the bottom of the upper end fitting and the top of the upper spacer grid.

A quick disconnect mechanism utilized on the latest version of the Mark-BZ fuel design, i.e., Mark-B10, is also used for the Mark-B11 fuel assembly. The attachments at the guide tube/upper end fitting interface allow the upper end fitting to be removed for fuel assembly reconstitution. The Mark-B10 cruciform leaf spring design, consisting of multiple leaf Inconel 718 material, is also utilized on the Mark-B11 assembly. Located in the upper end fitting, the spring maintains positive fuel assembly contact with the core support structure under all normal operating conditions and also maintains positive holddown margin for the Mark-B11 hydraulic forces.

All key dimensions are maintained to ensure compatibility with existing interfaces. All of the Mark-B11 features common to earlier Mark-BZ designs have been proven through extensive operational experience.

3.1.1.2 Fuel Rod

As with the previous Mark-BZ designs, the Mark-B11 fuel rod assembly comprises a Zircaloy clad fuel stack with Zircaloy end caps. The fuel rod cladding is a cold-worked, seamless, low tin, zirconium alloy. The Zircaloy upper and lower end cap designs are fundamentally unchanged from previous Mark-B designs. The upper end cap has a grippable notch to facilitate reconstitution and the lower end cap is bullet nosed and debris resistant, extending through the bottom end grid.

The fuel stack contains three zones: a central portion of enriched sintered uranium dioxide pellets and an axial blanket region at each end of the stack. The axial blanket region consists of sintered uranium dioxide pellets with a U^{235} enrichment of a low weight percent.

The fuel rod spring system employs one preloaded stainless steel spring in the upper plenum region that prevents movement of the fuel stack when subjected to shipping and handling loads. The fuel stack is seated on the lower end cap.

Other features of the fuel rod assembly are consistent with the fuel rod design changes previously incorporated into the Mark-B10 fuel rod design. These changes include a reduction in the pellet to cladding diametral gap from [b,c,d] inch to [b,c,d] inch and the removal of the lower plenum spring. The Mark-B10 fuel rods have been supplied to all three Oconee Nuclear Units starting with Unit 3, cycle 16 and have operated free of failures.

3.1.2 Unique Design Features

The specific Mark-B11 fuel assembly design features that enhance the nuclear, thermal-hydraulic, and mechanical performance include the following:

1. Reduced diameter fuel rod,
2. Flow mixing vanes on five of the six intermediate grid assemblies, and
3. Improved grid restraint system.

These features have been thoroughly evaluated analytically and empirically to ensure sufficient design margins and to confirm acceptable performance for batch implementation.

3.1.2.1 Fuel Rod

The most significant difference between the Mark-B11 fuel rod and its Mark-B predecessors is the reduction in the outer diameter from .430 inch to .416 inch. The 0.416 inch-diameter Mark-B11 fuel rod is configured in the same 15x15 array as the 0.430 inch-diameter Mark-B fuel rods. Using the same lattice, more water is contained within the boundary of the Mark-B11 fuel assembly, producing a softer neutron spectrum and a more neutronically reactive design. The softer neutron spectrum better utilizes the residual fissionable material in the adjacent 0.430 inch-diameter fuel rods. This added efficiency lowers enrichment costs for the fresh Mark-B11 fuel in transition cycles. In addition to large transition-cycle savings, the Mark-B11 design inherently requires lower boric-acid concentrations, which further reduces both operating costs and fuel-corrosion concerns.

Table 3.1 provides a comparison of Mark-B11 and Mark-B10 fuel rod parameters.

3.1.2.2 Flow Mixing Intermediate Grids

The Mark-B11 spacer grids are a direct evolution of Mark-BZ spacer grids. As with the Mark-BZ, upper and lower end grids are made of Inconel 718 strip material. The six intermediate grids are built from fully annealed, low-tin Zircaloy-4 and provide a fully keyable geometry to allow scratch-free and stress-free fuel rod insertion.

Unique Mark-B11 grid features include a reduction in fuel rod cell size (hard stop to soft stop) to accommodate the smaller diameter fuel rods and the addition of flow mixing vanes on the upper five intermediate grids. The cell size reduction ensures that the resulting fuel rod slip load remains unchanged. As shown in Figure 3.3, the mixing vanes maintain a conventional tab geometry on top of the spacer grid interior strips that bend outward from the plane of the strip. The vaned intermediate spacer grids provide improved thermal hydraulic performance by locally increasing the intensity of turbulence of the reactor coolant within the subchannel. Mixing vanes are not used on the lowermost intermediate spacer grids since the mixing enhancement is not necessary for this cooler region of the assembly.

Table 3.2 provides a comparison of Mark-B11 and Mark-BZ grid parameters.

3.1.2.3 Improved Grid Restraint System

As with previous Mark-BZ fuel assemblies and all FCF fuel designs, the intermediate grids are not fixed to the guide tube or instrument tube to help reduce fuel rod and fuel assembly bow. The grid restraint system allows the intermediate spacer grids to follow the fuel rods as they grow due to irradiation until the Zircaloy grids relax. After the spacer grids relax, intermediate grid axial motion is restrained through spacer grid inserts that contact cylindrical sleeves on the instrument tube.

The Mark-B11 design incorporates recent strength improvements made to the grid-to-sleeve interface on Mark-BZ fuel assemblies. Restraint sleeve-to-spacer grid interface geometries have been modified to increase strength. In addition, grid restraint load path improvements have been made on the Mark-B11 that in effect isolate the hydraulic loads for the two lowermost intermediate grids from that of the four uppermost grids.

The restraint sleeves are located between each spacer grid such that the hydraulic lift loads are transmitted through the top end grid for the upper four intermediate grids and through the bottom end grid for the lower two intermediate grids. This load path improvement serves to lower the load in the uppermost sleeves, which experience an increased hydraulic resistance attributed to the mixing vane grids.

Figure 3.1 - Mark-B11 Fuel Assembly

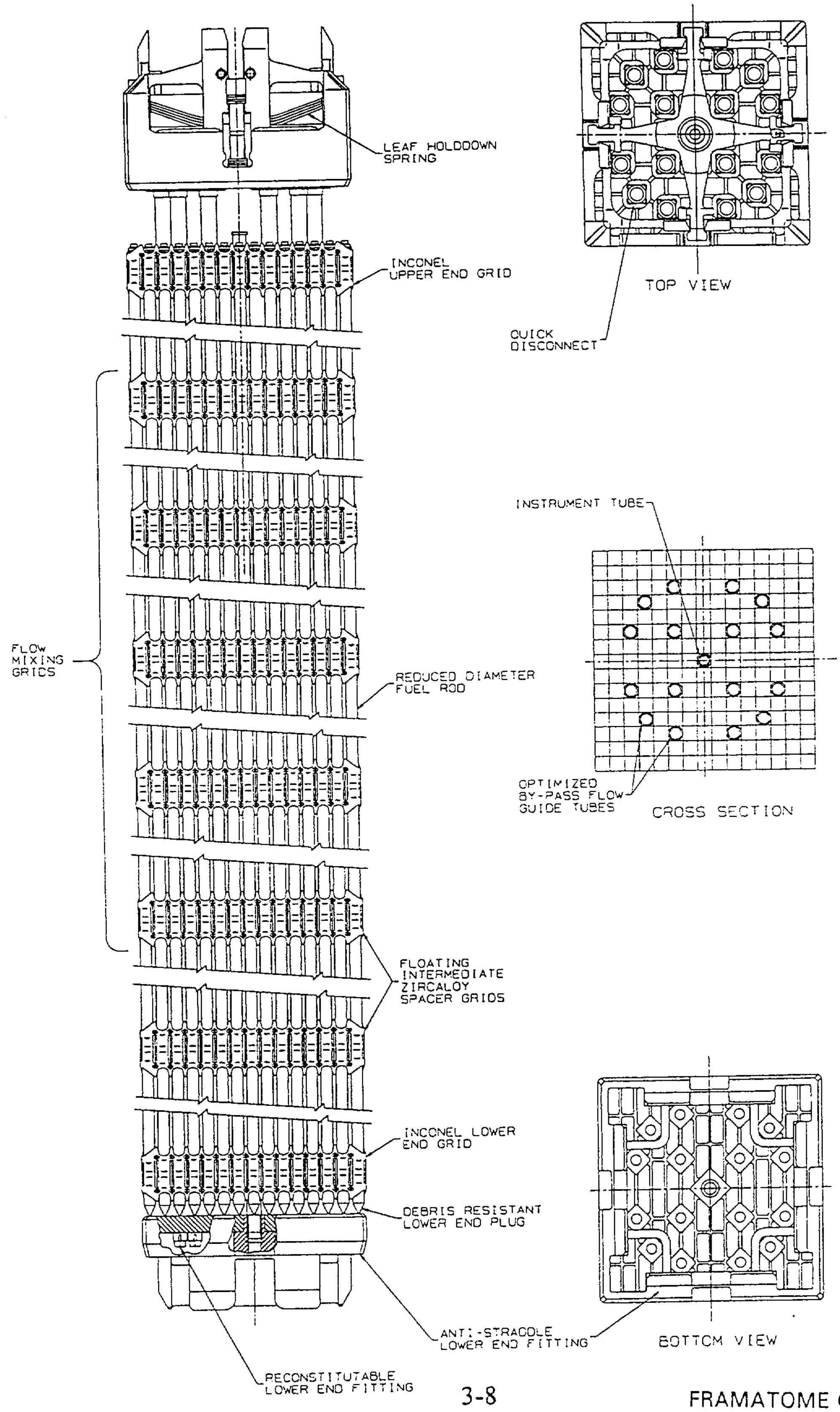

Figure 3.2 - Mark-B11 Fuel Rod Assembly

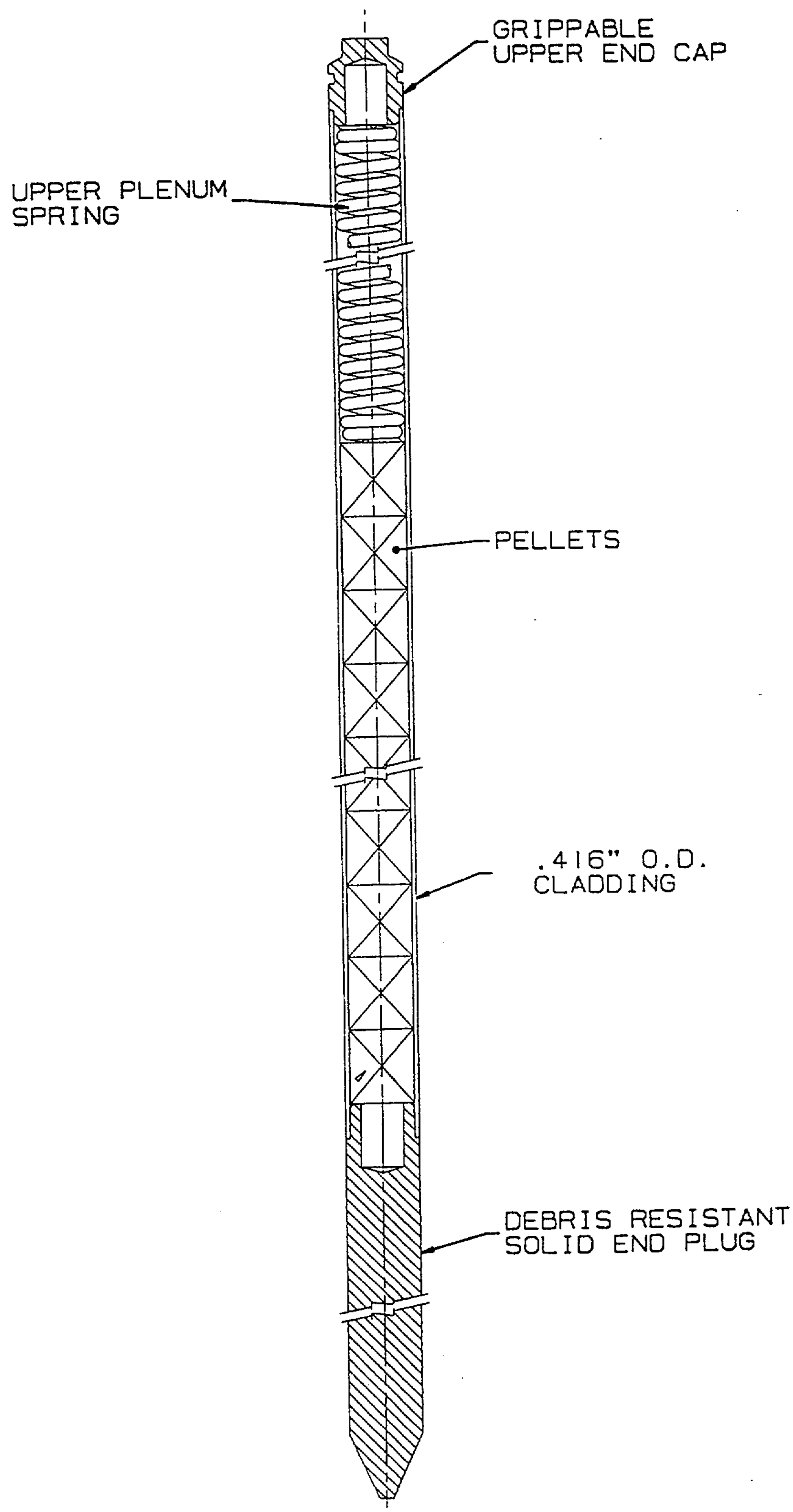

Figure 3.3 - Mark-B11 Mixing Vane Grid

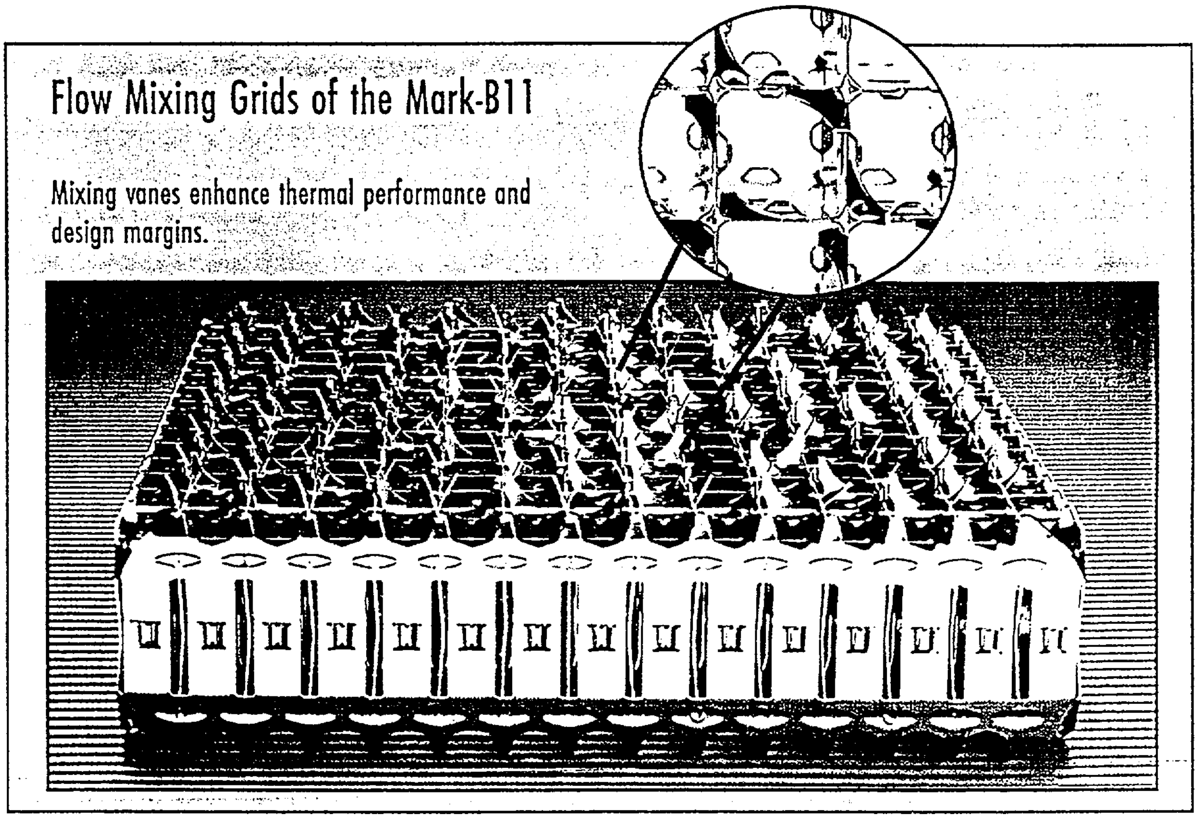

Table 3.1

Comparison of Mark-B11 and Mark-B10 Fuel Rod Parameters

Fuel Rod Parameters	Mark-B11	Mark-B10
Clad Material	Cold-Worked Stress Relieved Low-Tin Zircaloy-4	Cold-Worked Stress Relieved Low-Tin Zircaloy-4
Fuel Rod Length, in.	[b,c,d]	
Cladding OD, in.		
Cladding Thickness, in.		
Cladding ID, in.		
Clad-to-Pellet Gap, in.		
Fuel Pellet OD, in.		

Table 3.2

Comparison of Mark-B11 and Mark-BZ Grid Parameters

Grid Parameter	Mark-B11	Mark-BZ
Intermediate Grid		
Material	Fully Annealed Recrystallized Low-Tin Zircaloy-4	Fully Annealed Recrystallized Low-Tin Zircaloy-4
Mixing Vanes	Upper 5 Grids	N/A
Outer Strip Height, in.	[b,c,d]	
Outer Strip Thickness, in.		
Inner Strip Height, in.		
Inner Strip Thickness, in.		
Grid Envelope, in.		
Effective Cell Size, in.		
End Grid		
Material	Inconel 718	Inconel 718
Outer Strip Height, in.	[b,c,d]	
Outer Strip Thickness, in.		
Inner Strip Height, in.		
Inner Strip Thickness, in.		
Grid Envelope, in.		
Effective Cell Size, in.		

4. FUEL ASSEMBLY TEST PROGRAM

The Mark-B11 fuel design was subjected to a comprehensive test program to verify and characterize the mechanical and thermal-hydraulic performance. All testing addressed the key factors associated with the incorporation of the new Mark-B11 design features. Verification testing was conducted at various facilities. Critical heat flux testing was conducted at Columbia University in New York. Fuel assembly flow-induced vibration and pressure drop tests were performed in the Transportable Flow Test Rig (TFTR) at the Lynchburg Manufacturing Facility (LMF) using a full scale prototype. Additional pressure drop testing in addition to life and wear testing was performed at representative reactor conditions in the Control Rod Drive Line (CRDL) facility at the Alliance Research Center (ARC) in Ohio. Fuel assembly, spacer grid, and assembly component mechanical testing was performed at the LMF and ARC facilities. Results of Mark-B11 tests are summarized in the following sections.

4.1 Design Verification Testing

4.1.1 Flow-Induced Vibration Testing

Extensive flow-induced vibration (FIV) testing was conducted in the Transportable Flow Test Rig (TFTR) at the LMF facility. The purpose of the test was to examine the vibrational response of the Mark-B11 fuel assembly and to verify that no flow related phenomena existed that would adversely affect fuel integrity. The full-scale prototype testing also included the reactor-proven Mark-B10 fuel assembly to establish a baseline vibrational response for comparison to the Mark-B11 prototype. Testing was performed at low temperature and pressure conditions. Both assemblies were tested under a wide range of flow conditions, totaling more than 150 discrete flow intervals ranging from[b,c] to [b,c] gpm flowrate. Data analyses of 23 discrete parameters comprised more than 2500 different plots, characterizing detailed evaluations of the fuel assembly amplitudes and associated mode shapes as a function of flow rate.

As expected, neither the baseline Mark-B10 assembly nor the Mark-B11 prototype assembly exhibited any unusual resonant condition that would jeopardize fuel integrity. Both fuel assembly types were comparable in response. The observed resonances matched those calculated analytically. The amplitudes of vibration for both assemblies were very low with amplitudes less than[b,c,d]inch for a given frequency. Vibrational peaks that did appear were predictable and well behaved. Therefore, based on these test results and the life and wear test results (section 4.1.2), the Mark-B11 fuel assembly exhibits acceptable flow induced vibration performance under all reactor flow conditions. This has been further verified in that no operational problems or fuel failures have occurred in the Mark-B11 LTAs to date (section 4.1.3).

4.1.2 Life and Wear Testing

Life and wear testing of the Mark-B11 fuel assembly was conducted in the Alliance Research Center Control Rod Drive Line (CRDL) facility. The full-scale prototype assembly was subjected to 1,000 hours of endurance testing at simulated full power reactor operating conditions of temperature, pressure, flow, and coolant chemistry. The prototype assembly was constructed to simulate end-of-life (EOL) relaxed grid condition, which minimized the fuel rod-to-grid grip loads. The EOL condition is considered the most conservative to evaluate the effects of flow-induced fretting wear.

Post test inspections included detailed examination of fuel rods, spacer grids, guide tubes, the holddown spring, and the quick disconnect mechanism in the upper end fitting assembly. Component inspections revealed no indications of unacceptable wear. Fuel rod spacer grid contact wear was less than that of previous Mark-B fuel assembly designs for the same test conditions. Guide tube control rod wear was similar to that seen on previous Mark-B designs.

4.1.3 Lead Test Assembly Program

Final in-core verification is ongoing with the operation of four Mark-B11 LTAs at Oconee 2 Cycle 16, which began operation in April 1996. Given industry fretting problems associated with new fuel designs, the primary focus of the Mark-B11 LTA program is to ensure that the Mark-B11 fuel assembly is not subject to unexpected fuel rod/grid fretting failures. Three cycles of operation are currently planned. The first cycle of operation locates the Mark-B11 LTAs in the core interior, subjecting the assemblies to aggressive peaking values and verifying the interface with the burnable poison rod assembly (BPRA). The second cycle of operation locates the LTAs on the core periphery, subjecting the assemblies to baffle crossflow conditions, thus providing a bounding operating condition for flow-induced vibration and fuel rod fretting. The third cycle of operation relocates the LTAs in the core interior to maximize burnup while operating under a control rod assembly location. The Mark-B11 LTA program, coupled with the design verification testing and analyses and the proven experience of the Mark-BZ fuel assembly design at high burnups, serve to verify the Mark-B11 fuel assembly design for batch implementation.

4.2 Mechanical Tests

Extensive mechanical testing of the Mark-B11 fuel assembly and components was performed to provide input into analytical models and to demonstrate similitude with baseline Mark-B10/BZ fuel design. Testing consisted of fuel assembly mechanical testing including characterization of lateral and axial stiffness, natural frequency, structural damping; spacer grid impact testing; spacer grid static crush testing; grid restraint interface testing; and grid slip load testing.

4.2.1 Fuel Assembly Stiffness/Frequency

Mechanical testing was performed on the Mark-B11 fuel assembly to experimentally determine its lateral stiffness, axial stiffness, natural frequency, and damping characteristics. The prototype fuel assemblies represented end-of-life (EOL), simulating relaxed fuel rod slip load conditions. The relaxed condition represents that condition which exists for most of the fuel assembly design life. The results from these tests were used as inputs to benchmark the fuel assembly analytical models. The assembly was tested in air at room temperature in a special test fixture at the LMF facility. Testing consisted of dynamic pluck, axial stiffness, and lateral stiffness tests.

Table 4.1 provides the mechanical characteristics of the Mark-B11 and Mark-BZ fuel assemblies. Results show that the lateral and axial stiffness and natural frequency are within[b,c,d]for each of the two assemblies. Note that the Mark-BZ design tested was earlier design that utilized the lower end skirt, which effectively joined the lower end fitting to the lower end grid and stiffened the assembly slightly. Current Mark-BZ fuel designs do not utilize the skirt.

Table 4.1 - Summary of Mark-B11 Fuel Assembly Mechanical Test Results			
Characteristic	**Peak Deflection (In.)**	**Mark-B11 Fuel Assembly Results**	**Mark-BZ Fuel Assembly Results**
Lateral Stiffness (Lbs./In.)			
Axial Stiffness (Lbs./In.)		[b,c,d]	
Natural Frequency (Hz)			

* Lateral Stiffness was determined using a[b,c,d] axial preload. FA had bottom end skirt which increased stiffness.

** Lateral Stiffness was determined using a[b,c,d] axial preload.

4.2.2 Spacer Grid Impact Testing

Impact testing was conducted to determine the dynamic characteristics of the Mark-B11 intermediate spacer grids. These characteristics were used to determine inputs to the fuel assembly analytical models, to establish allowable impact loads, and to characterize the plastic deformation of the spacer grids.

Testing consisted of dynamic tests conducted at room temperature and at 600 °F. Table 4.2 includes test results of the Mark-B11 intermediate spacer grids in addition to those of the baseline Mark-BZ for comparison. The tests showed that no plastic deformation of the guide tubes occurred during the impact testing. The results showed that the strength and stiffness of the Mark-B11 intermediate grid compare favorably with the baseline Mark-BZ, resulting in higher average elastic impact force, average kinetic energy absorption, and damping while providing a slightly lower average stiffness. These results show that the Mark-B11 spacer grid increases structural margin.

Table 4.2 - Intermediate Spacer Grid Impact Test Results

Grid Type	Test Temperature	Average Elastic Impact Force (lbs)	Average Initial Kinetic Energy (in-lbs)	Average Stiffness (lbs/in)	Average Damping (ζeq or c/c_c)
Mark-B11	600 °F	[b,c,d]			
Mark-B11	~70 °F				
Mark-BZ	600 °F				

4.2.3 Spacer Grid Crush Test

Static crush testing was performed on Mark-B11 intermediate spacer grids to characterize spacer grid mechanical behavior for use in verifying shipping and handling loads. The static crush load for all of the spacer grids exceeded the required load capability of [b,c,d] pounds, which is derived from worst case shipping and handling loads.

4.2.4 Grid Restraint Interface Testing

Testing of the spacer grid restraint system was performed to determine the structural adequacy of the spacer grid to sleeve interfaces.

The spacer grid to sleeve interfaces were tested to failure at both cold and hot temperatures. All grid interfaces with restraining sleeves were tested to determine their load-carrying capacity under normal operation and faulted conditions.

Based on the positive margins obtained from each interface, the spacer grid to sleeve interfaces were shown to be structurally adequate for normal and faulted condition loads.

4.2.5 Spacer Grid Slip Testing

The purpose of the spacer grid slip testing was to measure the loads required to slip the spacer grids relative to the fuel rods, guide tubes, and instrument tube under ambient conditions. Results of this testing represent the total friction force between the spacer grids and the fuel rods and are used in the normal operating and shipping and handling analyses models. Slip load and load/deflection measurements were made for both the end and intermediate grids. The slip loads were within the expected range and were comparable to previous Mark-BZ baseline tests.

4.3 Hydraulic Tests

4.3.1 Pressure Drop Testing

Pressure drop testing of full-scale prototype Mark-B10 and Mark-B11 fuel assemblies was conducted in both the TFTR at the Lynchburg Manufacturing Facility and the CRDL facility at the Alliance Research Center. TFTR testing represented low temperature, pressure, and Reynolds number conditions. The CRDL testing represented in-reactor hot operating conditions at high Reynolds number conditions. The Mark-B10 testing served as a benchmark for comparison. Testing in the two test loops served to provide data for correlating the effects of Reynolds number. The pressure drop testing provided form loss coefficients for the Mark-B11 components, including the upper and lower end fittings, end grids, and intermediate grids, for input into thermal-hydraulic analyses discussed in sections 7.1 and 7.2. Excellent correlation of intermediate spacer grid form loss coefficients resulted between the TFTR and CRDL testing.

Component form loss coefficients for the Mark-B11 fuel assembly are provided in Table 4.3.

Table 4.3 - Mark-B11 Form Loss Coefficients

Mark-B11 Component	Form Loss Coefficients
Lower End Fitting	[b,c,d]
End Grids	
Non Mixing Grid	
Intermediate Mixing Grid	
Upper End Fitting	
Fuel Assembly	

4.3.2 Laser Doppler Velocimeter Testing

Extensive Laser Doppler Velocimeter (LDV) testing, conducted at the Virginia Military Institute Research Laboratories, provided a detailed description of the subchannel flow distribution within the Mark-B11 fuel assembly. The results from these tests were used to confirm the subchannel form loss coefficients, which were determined analytically, to establish the turbulent mixing coefficient used in thermal-hydraulic calculations and to ensure an acceptable velocity distribution.

The test apparatus consisted of a water flow loop, the test containment and the test rod bundle. Two test rod bundles were used. One consisted of a 5 x 5 section of fuel rods with a control rod guide tube in the center. The other consisted of four 3 x 3 fuel rod mini-bundles which simulated the corner regions of four adjacent Mark-B11 assemblies. All rod bundles were approximately[c,d]inches tall and contained three spacer grids each.

In order to characterize the velocity field of the coolant flow, velocity measurements were taken between the second and third grid. Measurements were taken along parallel lines through the subchannels at four cross-sectional planes.

The results of the two tests showed that the analytical subchannel form loss predictions could be correlated to the test results. In addition, no areas of flow starvation were found. The turbulent mixing coefficient for use in thermal hydraulic calculations is [b,c,d]which is the same value as used for similar FCF mixing grid designs.

4.3.3 Critical Heat Flux Testing

Critical heat flux (CHF) testing was conducted at Columbia University's Heat Transfer Research Facility. Testing conditions covered the full range of PWR operating conditions. A 5x5 array was tested using the mixing vane pattern from the Mark-B11 intermediate mixing grid design, which is a scaled version of FCF's Mark-BW17 design. The results of this testing showed that the BWCMV CHF correlation, originally developed for the Mark-BW17 design and documented in BAW-10159P-A [12], conservatively predicted CHF for the Mark-B11 fuel assembly design. CHF performance of the Mark-B11 assembly exceeded the BWCMV predicted performance by more than[b,c,d].

Further testing was conducted to quantify the CHF capability of the Mark-B11 grid. In all, 5 tests representing 3 different geometrical configurations were run. In BAW-10199P-A [13], a new CHF correlation form (BWU) was developed and a separate version was qualified for use with several grid designs. The version qualified for use with the Mark-B11 is termed the BWU-Z. The approved form of the BWU-Z CHF correlation, its associated design limit and its applicable independent variable ranges (as documented in Addendum 1 to reference 13) will be used in all analyses of the Mark-B11 mixing grid.

5. FUEL ASSEMBLY MECHANICAL EVALUATION

The Mark-B11 fuel assembly mechanical design criteria comply with that specified in BAW-10179, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" [2], which has been approved by the NRC. The fuel assembly design criteria ensure that the Mark-B11 fuel assembly, with the maximum credible damage, provides a path adequate for control rod insertion, maintains a coolable fuel rod geometry, and provides fuel assembly dimensions which remain within operational limits. Compliance with the criteria and methods identified in Reference 2 are discussed in the following sections.

The fuel assembly mechanical evaluation is divided into the following categories: growth, holddown, normal operation, faulted conditions (horizontal and vertical), fretting, fuel rod bow, shipping and handling, fuel assembly compatibility, material compatibility, and extended burnup. The fuel rod mechanical evaluation is considered separately from the fuel assembly and is addressed in section 6. Results of the analyses are applicable to fuel assembly operation in all Babcock & Wilcox-designed 177 fuel assembly skirt supported plants, including Duke Power Company's Oconee Nuclear Units 1, 2, and 3.

5.1 Fuel Assembly Growth

The Mark-B11 growth analysis conservatively predicts the maximum fuel assembly growth based on a statistical model assembled from Mark-BZ and Mark-BW post irradiation examination data. Using the minimum fuel assembly growth allowance and maximum upper confidence growth limit, the limiting fuel assembly burnup based on assembly growth is[b,c,d] MWd/mtU.

The Mark-BZ fuel assembly growth model, which includes assembly burnups as high as [b,c,d] MWd/mtU, is applicable to the Mark-B11 fuel assembly since the design changes implemented for the Mark-B11 fuel assembly will not affect assembly growth.

The Mark-B11 fuel design maintains the inherent FCF fuel design features of floating intermediate grids and seated fuel rods. This maintains the fuel assembly structural cage load paths and guide tube loads, which influence fuel assembly growth. The holddown spring remains unchanged and the fuel rod/spacer grid slip loads are comparable between the Mark-B10 and Mark-B11 designs. The reduction in fuel rod diameter is accommodated in the grid design as discussed in section 3.1.2.2, thereby ensuring the same fuel rod slip loads. The guide tube and fuel rod clad materials also remain the same as with earlier Mark-BZ designs. Given comparable axial loads and the same materials, the Mark-B11 fuel assembly growth will remain the same as that experienced in previous Mark-BZ designs.

5.2 Holddown

The evaluation of the Mark-B11 fuel assembly holddown capability ensures fuel assembly contact with the lower support plate during Condition I and II events. The fuel assembly upper and lower end fittings maintain engagement with reactor internals for all Condition I through IV events. The fuel assembly does not compress the hold down spring to solid height for any Condition I or II event. Mark-B11 holddown spring maximum loads and stresses are enveloped by bounding conditions evaluated for the Mark-B10 fuel application, therefore functional requirements are ensured.

The predicted lift loads are based on the Mark-B11 form loss coefficients listed in Table 4.3.1 and described in section 4.3.1. Sufficient holddown margin to prevent lift is provided. The lift evaluation is discussed in section 7.2.

5.3 Normal Operation

5.3.1 Stress

Stress intensities for Mark-B11 fuel assembly components were shown to be less than those limits established in reference 2, which were based on ASME Code , Section III criteria [6].

Temperature conditions ranging from the fourth pump startup temperature of 300°F to the operating temperature of 579°F for an operating pressure of 2,200 psia were considered. Beginning-of-life (BOL) and end-of-life (EOL) conditions were also evaluated to consider the change in load paths and loads due to material relaxation. The following fuel assembly components were evaluated:

1) Grid Restraint Sleeves/Inserts,
2) Guide Tube Assembly Components,
3) Upper and Lower End Fittings,
4) Quick Disconnect Components, and
5) Holddown Spring Assembly/Retainer.

Positive margins were determined for all fuel assembly structural components, showing that the Mark-B11 fuel assembly is structurally adequate for normal operating conditions.

5.3.2 Buckling

Buckling of Mark-B11 guide tubes was shown not to occur for normal operation conditions. Allowable guide tube axial loads were determined per reference 2, which limits the guide tube span axial load based on mid-span deflection criteria such as not to affect control rod insertion or trip performance. Guide tube corrosion, tolerances, and temperature effects were considered. Positive margins to buckling were determined for all temperature and fuel assembly conditions.

5.4 Faulted Conditions

The design bases used to establish the acceptance criteria for the Mark-B11 fuel assembly are provided in reference 2 and are consistent with NUREG-0800, Section 4.2, Appendix A [5] and follow the guidelines established by Section III of the ASME Code [6]. The design requirements for each category are as follows:

1) Operational Base Earthquake (OBE) - Allow continued safe operation of the fuel assembly following an OBE event by ensuring the fuel assembly components do not violate their dimensional requirements.

2) Safe Shutdown Earthquake (SSE) - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies, control rod insertibility, and a coolable geometry within the deformation limits consistent with the Emergency Core Cooling System (ECCS) and safety analysis.

3) Loss of Coolant Accident (LOCA) or LOCA Plus SSE - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the ECCS and safety analysis.

The Mark-B11 faulted evaluation addresses both the vertical (LOCA) and horizontal (LOCA and seismic) effects. The axial faulted analyses methodology is consistent with that submitted and approved by the NRC in BAW-10133P, Rev. 1 [4]. The horizontal faulted analysis methodology is consistent with that submitted and approved by the NRC in BAW-2292P, Rev. 0 [3]. The results are applicable to all Babcock & Wilcox-designed 177 fuel assembly plants with a skirt-supported reactor vessel.

5.4.1 Horizontal Analysis

The horizontal component of the faulted analysis determines the structural integrity of the Mark-B11 fuel assembly in the horizontal direction. The following loading conditions were evaluated:

1) Operating Basis Earthquake,
2) Safe Shutdown Earthquake,
3) Loss of Coolant Accident (LOCA), and
4) Combined Seismic and LOCA Events.

5.4.1.1 Stress

Stress intensities for Mark-B11 fuel assembly components were shown to be less than those limits established in reference 2, which were based on ASME Code , Section III criteria [6]. Mark-B11 fuel assembly components evaluated included those listed in section 5.3.1.

5.4.1.2 Grids

The Mark-B11 grids were evaluated applying the approved criteria and methodology described in references 2 and 4. No crushing deformation of the spacer grids is allowed for Condition I and II events and the spacer grids are required to provide adequate support to maintain the fuel rods in a coolable configuration for Conditions I thru IV.

The Mark-B11 evaluation showed that the predicted grid impact loads remain below the elastic load limits for all conditions including Operating Basis Earthquake, Safe Shutdown Earthquake, LOCA and combined SSE and LOCA conditions. Core plate time history inputs were determined using leak-before-break (LBB) methodology consistent with the NRC approved topical reports BAW-1847, Rev. 1 [19,18] and BAW-1999, Rev.0 [20]. The LBB core plate time history inputs utilized in the Mark-B11 analyses are the same as those used in the NRC approved Mark-B Grid Deformation Topical Report BAW-2292, Rev.0 [3, 21]. Seismic time histories corresponded to bounding spectra for the B&W reactor vessel skirt-supported plants.

The maximum faulted loads and corresponding allowable loads are based on grid impact testing reported in section 4.2.2. Results provided in Table 5.1 show that the grids remain elastic for all loading conditions, therefore control rod insertability and a coolable geometry are maintained for the Mark-B11 grids.

Table 5.1 - Grid Impact Loads

Faulted Condition	OBE		SSE		SSE + LOCA	
Direction	X	Z	X	Z	X	Z
Predicted Maximum Grid Force (lbs)	[b,c,d]					
Allowable Grid Force (lbs)						

[1] Conservative since plastic deformation is allowed but elastic load limit is used

5.4.2 Vertical Analysis

The Mark-B11 fuel assembly was evaluated for the vertical LOCA condition per the methodology provided in reference 2 to ensure control rod insertion and to ensure that all fuel assembly component stress limits are not exceeded.

5.4.2.1 Stress

Stress intensities for Mark-B11 fuel assembly components were shown to be less than those limits established in reference 2, which were based on ASME Code , Section III criteria [6]. Mark-B11 fuel assembly components evaluated included those listed in section 5.3.1. Positive margins were determined for all components.

5.4.2.2 Buckling

Mark-B11 guide tube buckling was evaluated for vertical faulted conditions per reference 2, considering the effects of guide tube corrosion, tolerances, and temperature effects. Allowable guide tube axial loads were determined based on the material yield stress per reference 2. Positive margins to buckling were determined for all fuel assembly conditions.

5.5 Fretting

The Mark-B11 fuel assembly was shown to provide sufficient support to limit fuel rod vibration and cladding fretting wear. Sections 4.1.1 and 4.1.2 provide discussion of results from the life and wear and flow induced vibration tests, both of which showed the Mark-B11 fuel assembly vibrational response is acceptable in terms of cladding and guide tube wear.

5.6 Fuel Rod Bow

Fuel rod bowing is evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly.

Post irradiation examination of Mark-B11 assemblies will determine the rod bow characteristics of the assembly. Mark-B11 fuel rod bow however is not expected to differ significantly from that of other FCF fuel assembly designs based on the same arguments presented for fuel assembly growth in section 5.1. The Mark-B11 fuel assembly maintains a similitude with earlier Mark-BZ designs with generic FCF features, materials and comparable fuel assembly loads.

5.7 Fuel Assembly Shipping and Handling

The Mark-B11 fuel assembly was evaluated for the structural adequacy for shipping and handling loads per reference 2. The analysis addresses loads on the Zircaloy and

Inconel spacer grids, upper and lower end fittings, guide tube, and guide tube attachments.

Positive margins were predicted for all the fuel assembly components considered. Positive margin against grid crush was demonstrated for a maximum load of[b,c,d]lbs during shipment (including grid clamping load). The Mark-B11 spacer grids were shown to maintain sufficient grip loads on the fuel rods to prevent axial movement during axial shipping and handling of up to 4 Gs. Lateral loads of up to 6 Gs were shown not to cause setting of the spacer grid spring stops.

5.8 Fuel Assembly Compatibility

Mechanical compatibility of the Mark-B11 fuel assembly with the reactor internals, handling and storage equipment, and resident fuel assemblies is verified through the similarity of the design to previous Mark-B fuel assemblies. The Mark-B11 fuel assembly upper and lower end fittings, the fuel assembly height and fuel assembly envelope are the same as the Mark-B10 fuel assembly. The axial positioning of the spacer grids is also maintained to avoid hang up with adjacent resident fuel assemblies and to provide adequate lateral interfaces.

5.9 Material Compatibility

The materials used in the manufacture of the Mark-B11 fuel assembly and fuel rod are compatible with all other materials in the primary system. All core components will continue to meet their required function since the Mark-B11 fuel assembly introduces no new materials to the core. Redesigned components such as the grid restraint parts, fuel rod components, and spacer grid assemblies utilize materials used in previous

Mark-BZ fuel assembly components and have been proven with extensive reactor experience.

5.10 Extended Burnup

All design and operational criteria are the same for extended burnup Mark-B11 fuel assemblies as for the original Mark-B fuel designs. The Mark-B11 fuel assembly will maintain its mechanical integrity at high burnups based on the existing FCF fuel database and the Mark-B11 similitude with previous fuel designs in addition to the extensive design verification program performed to date.

Extended burnup operation of the Mark-B11 fuel assembly is supported by a comprehensive series of post irradiation examinations carried out on previous Mark-B lead test assemblies, demonstration assemblies, and production fuel assemblies. As discussed earlier, similitude between the Mark-B11 and other FCF fuel assembly designs ensure satisfactory operation at extended burnups. Use of common reactor proven features, materials, components, design conditions and loadings, models, and mechanical characteristics allow for application of the data presented in BAW-10186P-A [1] to the Mark-B11 fuel assembly. Further confirmation will be made through post irradiation examinations (PIE) of the Mark-B11 lead assemblies. Examinations are scheduled to be conducted after the first and second cycles of operation in Oconee Unit 2. Key parameters will be measured and benchmarked to the data presented in reference 1. Additional PIE will be performed as required in future cycles to ensure sufficient monitoring of the Mark-B11 operational performance.

6. FUEL ROD MECHANICAL EVALUATION

The Mark-B11 fuel rod mechanical design criteria comply with those specified in BAW-10179 [2], which has been approved by the NRC. The mechanical evaluation demonstrated the structural integrity of the Mark-B11 fuel rod design. The evaluation addresses the following areas of mechanical performance: corrosion, creep collapse, transient strain, stress, fatigue, shipping and handling, fuel rod growth, and fuel rod fretting. Each of these areas is discussed in the following sections. All of the following fuel rod mechanical evaluations represent generic values, which would generally be more conservative than cycle specific analyses. Thus, the reported results should be treated as typical values. Cycle specific analyses using the same approved methods and models would be the analysis basis for each cycle.

6.1 Corrosion

Oxide layer growth on the fuel rod cladding surface inhibits several areas of mechanical performance. During the corrosion process, base metal converts to oxide, reducing the effective thickness of the Zircaloy. The cladding also operates at higher temperatures due to the lower thermal conductivity of the oxide relative to the base metal. For this reason, a conservative oxide layer thickness of [b,c,d] is assumed to be present on the cladding outer surface in the cladding stress and fatigue analyses. Further, cladding outer surface oxide thickness is predicted for comparison to a steady state operating limit of [b,c,d]. This limit and the model used to predict FCF cladding corrosion are documented in BAW-10186P-A [1]. The corrosion model is licensed to predict FCF cladding corrosion performance to a fuel rod average burnup of[b,c,d] MWd/mtU. Mark-B11 fuel rod corrosion analyses utilize the models and corrosion limit set forth in reference 1, using conservative cycle specific radial power history and axial flux shapes.

6.2 Cladding Transient Strain

Transient strain occurs as a result of cladding deformation caused by fuel pellet radial swelling during power increases. Uniform transient strain, both elastic and inelastic is limited to 1.0%. BAW-10186P-A [1] contains the transient strain analysis methodology.

The transient strain analysis results in a local fuel rod linear heat rate versus rod average burnup limit that prevents the fuel rod from achieving 1.0% strain. For the Mark-B11 fuel rod, the generic local linear heat rate limit remains above [b,c,d] to a rod average burnup of [b,c,d] MWd/mtU.

6.3 Cladding Stress

Reference 2 defines the FCF Mark-B cladding stress analysis methodology. Stress level intensities are calculated in accordance with the ASME Code, which includes both normal and shear stress effects. These stress intensities are compared to 2/3 of the minimum specified unirradiated yield strength of the material at operating temperature.

Using the reference 2 methodology, the Mark-B11 fuel rod was shown to maintain positive margins between the maximum predicted stress intensities and the allowable stress. The minimum generic margin is[c,d], achieved while combining primary membrane stresses predicted under normal and transient (non faulted) operating conditions.

6.4 Cladding Fatigue

During core operation of the fuel rod, various plant maneuvers cause power fluctuations, or transients, which can result in large pressure and temperature oscillations in the fuel rod and fuel rod cladding. These oscillations lead to fluctuating thermal, pressure and ovality stresses in the fuel rod cladding and can ultimately lead to fatigue failure. The cladding fatigue analysis models these transients using TACO3, FCF's fuel pin thermal analysis code described in BAW-10162P-A [8]. Reference 2 contains the FCF analysis methodology and criterion, which limits the total fatigue usage factor for all Condition I and II events to 0.9.

For the Mark-B11 fuel rod, individual utilization factors for each applicable transient were calculated and summed to find the total generic utilization factor of[b,c,d]. Since this is less than the 0.90 total allowable usage factor, the Mark-B11 fuel rod design is acceptable in terms of cladding fatigue up to a design life of 10 effective full power years.

6.5 Creep Collapse

The FCF cladding creep collapse analysis methodology and corresponding CROV computer code are established in BAW-10084P-A [7] and approved to a rod average burnup of[b,c,d] MWd/mtU per reference 1. Creep collapse of the cladding due to creep ovalization shall not occur during the in-core life of the fuel rod. Predicted creep collapse occurs when the creep ovalization rate exceeds 0.1 mils/hour or the maximum fiber stress exceeds the unirradiated yield strength of the cladding.

Both TACO3 and CROV codes were used to model the Mark-B11 fuel rod in core cladding creep performance. Analytical results show that Mark-B11 fuel rod creep collapse life exceeds[b,c,d] effective full power hours, which is equivalent to a burnup of 70,000 MWd/mtU for the power history analyzed. Use of a less restrictive power history would result in a longer creep collapse life in terms of hours.

6.6 Fuel Rod Growth

The gap allowance between the upper end fitting and the fuel rod assembly is designed to provide a positive clearance during the assembly lifetime. Reference 1 contains the analysis methodology and shows that it is approved to a rod average burnup [b,c,d] MWd/mtU.

The fuel rod growth model is based on FCF irradiation experience obtained with cladding material equivalent to that of the Mark-B11. The model predicts the fluence at which the gap closes. The predicted fluence is then related to a rod average burnup. The gap prediction uses the upper tolerance limit model for fuel rod growth in Mark-BZ type fuel assemblies and the lower tolerance limit model for assembly growth. Results for the Mark-B11 show that a positive gap is maintained at a rod average burnup greater than[b,c,d] MWd/mtU.

6.7 Shipping and Handling

Per reference 2, the Mark-B11 fuel rod spring system must be able to withstand a 4G axial loading from the fuel stack mass during shipping and handling without any gaps larger than[b,c,d] inch forming within the fuel rod internals. Mark-B11 fuel rod analyses demonstrated that this criterion is met using the approved method of reference 2.

6.8 Fuel Rod Reliability

The reliability of the Mark-B11 fuel design is expected to be excellent. The reliability of all FCF fuel designs has improved over the last few years to where all FCF fuel is now operating leaker free. This improvement was based on a comprehensive program to review and improve critical design and fabrication parameters. The Mark-B11 fuel design shares these same proven parameters from both the Mark-B and Mark-BW product lines. The Mark-B11 fuel is expected to have the same excellent fuel reliability as demonstarted by the successful design verification testing and lead test assembly operation to date.

The 0.416 inch-diameter Mark-B11 fuel rod was designed using similar parametric relationships as the proven Mark-B and Mark-BW fuel rod designs. It also has similar margins to mechanical design criteria. FCF has fabricated fuel rods with outside diameters of 0.430, 0.422, and 0.374 inch. Fabrication will be made with the same manufacturing equipment and processes used to fabricate the Mark-B and Mark-BW fuel rods.

The Mark-B11 spacer grids are similar to those used in the Mark-BZ and Mark-BW fuel designs. Mark-BZ, including Mark-B11, and Mark-BW spacer grids are similar in cell construction (hard and soft stop configuration). The Mark-B11 utilizes a similar mixing vane geometry and pattern as the Mark-BW design. Both the Mark-B11 and Mark-BW use mixing vanes on the uppermost five intermediate spacer grids. Life and wear and flow-induced vibration testing of the Mark-B11 design, using simulated end of life grid conditions, showed acceptable spacer grid to fuel rod wear and fuel assembly dynamic response under reactor flow conditions. In core operation of the Mark-B11 lead test assemblies has also shown good performance with no problems experienced.

7. FUEL ASSEMBLY THERMAL-HYDRAULIC EVALUATION

7.1 Core Pressure Drop

As described in Section 4.3.1, the pressure drop characteristics of the Mark-B11 fuel assembly were determined through a series of flow tests at the Alliance Research Center and at the Lynchburg Manufacturing Facility. The results of these tests were used as the basis for the calculation of component form loss coefficients for the end fittings and spacer grids.

Analyses were performed using the NRC approved LYNXT code per BAW-10156-A [14] to establish pressure drop characteristics of the Mark-B11 fuel assembly in full core and mixed core implementation with resident non-mixing grid fuel. The mixed core analyses compared the overall pressure drop of the Mark-B11 and Mark-B10 assemblies as well as the pressure drop of individual components. The pressure drop of the Mark-B11 is lower than the Mark-B10 up to the first mixing grid due to the smaller fuel rod diameter thereby creating a flow diversion into the Mark-B11. At the mixing grid locations flow is diverted back into the surrounding Mark-B10 assemblies. In the spans between mixing grids, flow returns back to the Mark-B11. Even with these flow diversions, the crossflow velocity is less than the[b,c,d] maximum crossflow criterion. Up to the first mixing grid, the Mark-B11 pressure drop is[b,c,d] than the Mark-B10. At each mixing grid, the Mark-B11 pressure drop is [b,c,d] than the non-mixing grid. The smaller fuel rod diameter results in lower friction pressure drop, which helps offset the increased pressure drop of the Mark-B11 mixing grids. Overall, the Mark-B11 pressure drop is [b,c,d] than the Mark-B10. This close matching of the Mark-B11 and Mark-B10 overall pressure drop ensures that there will be no adverse impact on hydraulic lift loads, core internals loading, RCS flow rate, core bypass flow rate, and control rod drop times.

7.2 Fuel Assembly Hydraulic Lift

The hydraulic lift force on a fuel assembly is attributed to the component and friction pressure drop across the length of the assembly. Extensive hydraulic testing on the Mark-B and Mark-BW series of fuel has included many hydraulic lift tests at the Alliance Research Center at in reactor conditions. Based on this testing, it has been observed that for all components, except the top nozzle, the hydraulic lift formloss coefficients are equal to the component pressure drop formloss coefficients. A method to adjust the upper end fitting form loss coefficient has been derived to match the analytical lift predictions to the lift experiments and this has been applied to the Mark-B11 fuel assembly.

Using the calculated form loss coefficients and the LYNXT code, along with bounding assumptions on inlet flow conditions, several analyses were performed which evaluated the hydraulic lift forces on the Mark-B11 in both a full core and in a mixed core environment. The mixed core analysis showed that the total lift force on the limiting Mark-B11 fuel assembly would be[b,c,d] than the limiting assembly in a full Mark-B10 core. A full core Mark-B11 analysis shows that the lift force on the Mark-B11 is [b,c,d] than in a full Mark-B10 core. Therefore, the Mark-B11 fuel lift loads are bounded by the Mark-B10 values.

7.3 Core DNB

The purpose of the core DNB analysis is to insure that there is a 95% probability, with a 95% confidence that no fuel rod will experience a departure from nucleate boiling (DNB) during normal operation or transients of moderate frequency (reference 2). The Mark-B11 fuel assembly implements two design evolutions that affect DNB performance. Mixing grids improve DNB performance and the slightly smaller fuel rod diameter decreases DNB performance relative to the Mark-B10. The impact of these competing design changes was evaluated using the LYNXT cross flow code with variable transverse scaling. The BWU-Z CHF correlation was used in the Mark-B11 LYNXT analyses as described in section 4.3.3.

Core thermal hydraulic analyses performed to demonstrate that the DNB criterion is met use a reference design power distribution, called "design peaking", that is assumed to bound, in terms of DNB performance, real power distributions occurring during plant operation. To provide assurance that this assumption is valid, maximum allowable peaking (MAP) limits are developed. These limits are a family of curves for which the minimum DNB ratio (MDNBR) is equal to a target value, typically the DNB analysis limit. The MAP limits provide linkage between the DNB analyses and the core operating and safety power distribution limits.

For the 177 fuel assembly B&W reactors, MAP limits are developed at the RCS DNB safety limit statepoints and the limiting statepoint from the most limiting loss-of-coolant flow transient. The first set is called the reactor protection system (RPS) MAPs and the second is referred to as the operating limit (OL) MAPs. The impact on both types of MAPs due to the implementation of the Mark-B11 fuel assembly in a full and mixed core configuration has been evaluated.

7.3.1 Steady-State DNBR

The effect on design thermal margins during steady state operation is evaluated at a constant power level (the maximum achievable steady-state power, or design overpower condition). RPS MAP limits were generated for both a full core of Mark-B10 and Mark-B11 fuel using a traditional treatment of uncertainties. Alternatively, the Statistical Core Design (SCD) technique could be used for this comparison providing similar indications of performance change on a relative basis. For the Mark-B10 fuel the BWC CHF correlation per BAW-10143P-A [15] was used and the MAPs to reach the BWC MDNBR design limit of [c,d] were determined. The BWU-Z CHF correlation was used and adjusted by the[c,d] multiplier for the Mark-B11 fuel and the MAPs to reach the Mark-B11 BWU-Z design limit of [c,d] were calculated then compared to the Mark-B10 MAPs. This comparison shows that the Mark-B11 provides at least [c,d]and up to [b,c] peaking margin depending on the axial peak / elevation combination. In terms of DNB margin this equates to at least [c,d] and up to[c,d] additional margin to the DNB analysis limit.

The limiting mixed Mark-B10/Mark-B11 core is a single Mark-B11 assembly in a Mark-B10 core. This configuration maximizes the effects of flow diversion at the Mark-B11 mixing grid locations and any associated DNB penalty. The same process as followed in the full core analysis was used and the calculation showed that the Mark-B11 DNB performance remains significantly above the Mark-B10 at between[b,c,d]and [b,c,d] increase in MAP limits. However, relative to the full core Mark-B11 RPS MAPs the mixed core values are in some cases higher and in some cases smaller. The maximum variation occurs for the[c,d]axial peak with the maximum increase being[b,c,d]at a normalized elevation (x/l) of[c,d]and the maximum decrease being[b,c,d] at an x/l of [b,c,d].

A comparison of the Mark-B11 and Mark-B10 designs shows that these results are expected. The Mark-B11 pressure drop is lower until the first mixing grid is encountered. Therefore, flow is being diverted at first into and then out of the Mark-B11. With the higher axial peaks, the point of MDNBR occurs closer to the point of maximum heat flux and, therefore, further down in the core. So, in the lower x/l cases there is a benefit in the mixed core configuration and as the point of MDNBR moves higher there is a penalty due to flow diversion out of the Mark-B11 assembly at the mixing grids.

7.3.2 Transient Analysis

The transient DNB analysis ensures that the 95/95 DNB criterion is met for transients of moderate frequency. The limiting, moderate frequency transient for 177 fuel assembly cores, in terms of DNB margin, is typically a partial loss of coolant flow transient such as a one or two pump coastdown. The statepoint analyzed is the steady state equivalent of the limiting time during the transient. The OL MAPs are determined in a manner similar to the RPS MAPs except that the DNB target is the MDNBR during the transient. The two pump coastdown was chosen for this margin comparison. Similar trends are expected for a one pump coastdown. The full and mixed core OL (transient) MAP analyses show that the Mark-B11 fuel provides increases in MAP margins greater than the RPS (steady state) MAP analyses. For the full core case the OL MAP margin increase is between[b,c,d] and[b,c,d] and for the mixed core case the increase is between[b,c,d] and [b,c,d].

All cases show positive MAP margin relative to the resident fuel thereby ensuring that the DNB criterion will be preserved during the transition to full core implementation of the Mark-B11 fuel design.

7.4 Fuel Rod Thermal-Hydraulic Evaluation

7.4.1 Fuel Rod Internal Pressure

The internal pressure of the peak fuel rod in the reactor is limited to a value below that which would cause the fuel-clad gap to increase due to outward cladding creep during steady-state operation thereby ensuring that extensive DNB propagation does not occur.

The Mark-B11 fuel rod internal gas pressure was determined using the TACO3 computer code per BAW-10162P-A [8] and the methodology defined in BAW-10183P-A [9]. The results show that the fuel rod can attain burnups in the range of GWd/mtU depending on the axial flux shapes used in the analysis. Inputs to the analysis include a power history that is assumed to envelop the operation of any individual fuel rod and worst case manufacturing variations allowed by the fuel rod specifications. Higher allowable burnups would be achieved on a cycle specifc basis by utilizing fuel rod specific power histories, fuel assembly as-built manufacturing data, and a more realistic total peak uncertainty.

7.4.2 Centerline Fuel Melt Limit

Fuel melting is not permitted during normal operating conditions or during anticipated operational occurrences. The TACO3 computer code was used to determine the local linear heat rate throughout the fuel rod lifetime that results in centerline temperature predictions exceeding T_L, a limit value chosen such that a 95% probability exists at the 95% confidence level that centerline melting will not occur. The most limiting time-in-life for the local linear heat rate is at the beginning of life. A typical generic centerline fuel melt limit is [b,c,d] for the Mark-B11 fuel rod.

8. NUCLEAR DESIGN EVALUATION

The Mark B-11 fuel assembly is similar to the Mark B-10 design from a neutronics viewpoint except that it has a smaller fuel rod diameter (0.416 inch vs 0.430 inch). Other changes, such as the design of the spacer grids, are minor.

The reduced fuel rod diameter results in a lower uranium loading and an increase in neutron moderation because of the added water in the fuel rod cell. On an equal enrichment basis, the Mark B-11 design initially exhibits greater reactivity than the Mark B-10 design. This difference diminishes with burnup and eventually it has less reactivity than the Mark B-10 design because the softer neutron spectrum resulting from the additional water in the cell has resulted in lower plutonium production. This behavioral difference has no adverse impact on the operation of the plant.

Shutdown margin is greater with the Mark B-11 design than with the Mark B-10, but this is not a significant factor because the plants that could utilize the Mark B-11 design all have more than sufficient control rod worth. Moderator coefficients are less negative throughout the cycle with the Mark B-11 design but well within the range normally encountered in reload designs. BOC moderator coefficients are easily controlled with burnable absorbers. A less negative EOC moderator coefficient is advantageous because of its beneficial effect on certain postulated accidents such as the steam line break and on shutdown margin. The Doppler coefficient is slightly less negative in the Mark B-11 design but within the range assumed in safety analyses.

From a physics viewpoint, the Mark B-11 assembly design does not present a large change from the Mark B-10 design and earlier designs already licensed and operated. The Mark B-11 can be used alone or in conjunction with the Mark B-10 or earlier designs without adversely affecting plant operation or safety.

9. ECCS EVALUATION

The Mark-B11 fuel assembly differs in design from other Mark-B fuel types such that its performance and coolability during a postulated LOCA must be evaluated. The two main differences affecting LOCA analyses are the change in the fuel pin outside diameter (OD) and mixing vane grids. The smaller pin OD reduces the surface area for heat transfer, while the mixing vane grids change the axial flow resistance. Analysis of the post-LOCA performance of the Mark-B11 fuel assembly has shown that it is in compliance with the five criteria of 10 CFR 50.46. The LOCA analyses were performed in accordance with the RELAP5/MOD2-based per BAW-10164P Rev. 3 [16] Evaluation Model (EM) described in BAW-10192P, Rev. 0 [17]. Analyses were performed for small and large LOCA scenarios. Noding and convergence sensitivity studies appropriate for each range of break sizes were also performed.

Two SBLOCA break spectrums were analyzed with Mark-B11 fuel using the BWU-Z CHF correlation. The first set of analyses postulated the LOCA from[c,d] percent full power and utilized two HPI pumps to mitigate the consequences of the SBLOCA. The second set of analyses was postulated from[b,c,d]percent power with one HPI pump and steam generator blowdown to augment the RCS depressurization rate.

Two sets of LBLOCA analyses were performed using the BWU-Z CHF correlation. The first set modeled an entire core of Mark-B11 fuel assemblies (whole-core). The second set modeled a core with Mark-B11 and Mark-B10, or hydraulically similar fuel assemblies, in mixed-core analyses. The increased resistance of the Mark-B11 mixing vane grid resulted in flow diversion out of the Mark-B11 assembly. Accordingly, any lead test assemblies or the first two full batches of Mark-B11 fuel incorporated into any core will have LBLOCA linear heat rate limits that are less than those calculated for the whole-core configuration.

10. DESIGN EVALUATION SUMMARY

The Mark-B11 fuel assembly was shown to meet all fuel assembly design criteria critical to safe and reliable operation. The features new to the Mark-B11 fuel design, which include the reduced diameter fuel rod, flow mixing vanes, and a redesigned grid restraint system, meet all fuel assembly mechanical, thermal-hydraulic, core physics, ECCS, and safety criteria. The standard Mark-BZ features maintained in the Mark-B11 assembly provide reactor proven design parameters that provide a basis for successful future performance. Design verification testing and analyses have demonstrated the acceptability of the added design features and ensure that Mark-B11 fuel assembly will operate safely and reliably. A detailed LTA program will further verifiy the Mark-B11 irradiation performance for benchmarking to existing models and data which have presently been defined as representative of the Mark-B11 design.

Acceptable Mark-B11 fuel assembly and fuel rod mechanical and thermal-hydraulic performance capability can be obtained for fuel rod average burnups up to [b,c,d] MWd/mtU. Therefore, FCF fully expects to utilize the burnups specified in BAW-10186P-A, "Extended Burnup Evaluation", and approved by the NRC, for the Mark-B11 fuel assembly design.

11. REFERENCES

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3. BAW-2292, Rev. 0, Mark-B Fuel Assembly Spacer Grid Deformation in B&W Designed 177 Fuel Assembly Plants, March 1997.

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10. BAW-10156-A, LYNXT, Core Transient Thermal-Hydraulic Program, August 1993

11. BAW-1829, Thermal Hydraulic Crossflow Application, April 1984.

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15. BAW-10143P-A, BWC Correlation of Critical Heat Flux, April 1985.

16. BAW-10164P-A Rev. 3, "RELAP5/MOD2-B&W, An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," July 1996.

17. BAW-10192P, "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," February 1994.

18. NRC Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, December 2, 1985.

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19. BAW-1847, Rev. 1, Leak-Before-Break Evaluations of margin Against Full Break for RCS Primary Piping of B&W Designed NSS, Babcock & Wilcox, Lynchburg, Virginia, September 1985.

20. BAW-1999, Rev. 0, TMI-1 Nuclear Power Plant Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping, Babcock & Wilcox, April 1987.

21. NRC Safety Evaluation Report of BAW-2292, David B. Matthews to J.H. Taylor, August 20, 1997.

APPENDIX A

Non-Proprietary Responses to Request for Additional Information for BAW-10229P, Mark-B11 Fuel Assembly Design Report

***Question 1:** Please provide a history of the evolution of the Mark B fuel designs in the last 10 years defining the new features introduced with each design.*

Response 1: Below is a summary of all Mark-B designs since the implementation of zircaloy intermediate grids. The earliest introduction of the zircaloy intermediate grid fuel assembly type was in 1982 in Oconee Unit 3. The latest introduction was in 1988 at TMI Unit 1 and ANO Unit 1. As a timeline, the year of introduction at the Oconee Nuclear Station is shown in the responses below.

B4Z Zircaloy Intermediate Grids, Annealed Guide Tubes (1982)
B5 Modified Upper End Fitting (UEF) and Holddown Spring Retainer, Anti-straddle Lower End Fitting (LEF) (1982)
B6 Reconstitutable UEF / Skirtless upper end grid (1986)
B7 Shorter LEF, shorter lower end grid skirt, reduced pre-pressure, longer fuel rod for higher burn-up (1987)
B8 Debris Resistant lower end cap (1990)
B9 Removable LEF (no skirt), redesigned lower grid restraint, larger diameter fuel pellet, rod pre-pressure optimized for modern fuel performance codes, low tin clad (1.2 –1.4%), optimized pellet/clad gap, optimized bypass flow guide tubes (1991)
B10 Cruciform Leaf Holddown spring (1991)
B10 options:
Quick disconnect UEF
Zone loading of multiple enrichments within one F/A
Axial blanket fuel rods
Increased diameter fuel pellets with thin wall cladding and axial blankets
Zone loading of Gadolinia fuel rods within one F/A
Fuel rod with increased plenum volume

***Question 2** On page 4-10, the report provides the departure from nucleate boiling ratio (DNBR) design limit for Mark-B11 fuel, and references Appendix E of BAW-10199P-A (which documents the form of the BWU-Z correlation applicable to Mark-B11 fuel). Appendix E of BAW-10199P-A has not been approved by the NRC, and is currently under review as Addendum 1 to BAW-10199P-A (which consists of Appendix E and Appendix F of BAW-10199P-A). Therefore, the Mark-B11 fuel assembly design topical report (BAW-10229P) does not contain an approved critical heat flux (CHF) correlation for this fuel design.*

a) If the approved form of the BWU-Z correlation is different from that originally submitted, how will these changes be incorporated into BAW-10229P to avoid *confusion over the proper design limit to use with Mark-B11 fuel assemblies in operating plants?*

b) What are the range of conditions that the approved BWU-Z correlation will be applied to, e.g., flow rate, pressure, temperature, etc., for Mark-B11 fuel assemblies? How are these ranges validated?

Response 2 The second paragraph on page 4-10 refers to the BWU-Z CHF correlation as applicable to the Mark-B11. The last two sentences of this paragraph will be deleted and replaced by:

"The approved form of the BWU-Z CHF correlation, its associated design limit and its applicable independent variable ranges (as documented in Addendum 1 to reference 13) will be used in all analyses of the Mark-B11 mixing grid."

To avoid confusion with other versions of BWU, the form approved for use with the Mark-B11 grid will be designated "BWU-B11".

As stated on page E-5 of Addendum 1 to BAW-10199P-A, the ranges of applicability for Mark-B11 application are specified in Table 3-1. These ranges are 0.36 to 3.55 million pounds per hour per square foot in mass velocity, 400 to 2465 psia in pressure and an equilibrium quality of less than 0.74 at the minimum CHF ratio. The design limits in Table 3-1 are a function of system pressure: 1.19 above 1000 psia 1.20 from 700 to 1000 psia and 1.59 below 700 psia.

These ranges were verified in a five test program described in detail in Appendix E of Addendum 1 to BAW-10199P-A. The ranges tested in this Mark-B11 test program were 0.377 to 3.095 million pounds per hour per square foot in mass velocity, 595 to 2425 psia in pressure and equilibrium qualities up to 0.6025 at the location of CHF. These values are shown in Table E-7, page E-15. The mass velocity and equilibrium quality ranges were equivalent to those of the original BWU-Z data base. While the lower pressure range was limited to 595 psia, both the 600 and 1000 psia groups performed above the average (Table E-7, Grouped by Pressure, page E-15). Thus the application of the higher (more conservative) design limits of Table 3-1 at the lower pressures is justified.

Question 3: *Please provide more detail describing the results of the laser doppler velocimeter (LDV) testing conducted at the Lynchburg manufacturing facility (LMF) to investigate subchannel flow distribution within the Mark-B11 fuel assembly? Specifically:*

a) What ranges of operating conditions (e.g., flow rate, pressure, temperature) were tested?

b) Provide plots of the measured versus predicted pressure drop in the various subchannels, for the full range of flows and pressures tested?

c) Provide a discussion of the comparison between the analytical model predictions and the measured data?

Response 3: The LDV test conducted at Virginia Military Institute Research Laboratory (VMIRL) obtained detailed velocity information on the mixing vane grid. [

e.

] The plots of the test results are shown on Figures 3-1 through 3-8.

The purpose of LDV testing was to determine the flow patterns induced by the vaned intermediate spacer grid. The GRIL code was used to analytically model the grid and to distribute the total grid form loss over the grid subchannels. The subchannel form loss coefficients were then used in the LYNXT code to predict the flow pattern. By comparing the LYNXT predictions with the LDV data, the adequacy of the GRIL subchannel form loss coefficients could be determined. The 5x5 LDV test results matched the LYNXT flow distribution with an average difference of [d]. The 6x6 model showed more discrepancy but still had an average error of less than [d]. These results are similar and consistent with previous fuel assembly designs.

Question 4: *Please provide more detail describing the pressure drop testing of the full-scale prototype B11 assembly. Specifically,*

a) What ranges of conditions were tested in the transportable flow test rig (TFTR) and the control rod drive line (CRDL)? How were the pressure drop measurements obtained? How were the form losses determined from the measured data?

b) What were the corresponding form losses in the Mark-B10 fuel assembly design? How were the Mark-B10 pressure drop measurements used as a *"benchmark" for the Mark-B11 measurements?*

Response 4: The pressure drop measurements were obtained in a manner consistent with previous fuel designs. Both the TFTR and CRDL test programs evaluated the individual component and overall pressure drop characteristics of a full-scale prototype fuel assembly. The two test programs differed in that the CRDL at the Alliance Research Center (ARC) provides flow, pressure and temperature conditions as seen in an operating reactor whereas the TFTR is a low pressure, low temperature, full flow facility. Hydraulic flow tests at the CRDL have included lift, pressure drop, life and wear, and RCCA trip tests. Only pressure drop and lift tests have been performed at the TFTR. A summary of the test conditions is provided below.

	TFTR	CRDL
Flow, gpm	[d.]	
Temperature, F	[d.]	
Pressure, psia	[d.]	

The TFTR and CRDL were equipped with instrumentation to measure loop flow parameters and fuel assembly component pressure drops. Differential pressure transmitters were used to measure the pressure drops across the fuel assembly spans and the flow nozzles. A pressure transmitter was used to measure the loop pressure below the fuel assembly. Thermocouples were used to measure loop temperatures, heat exchanger temperatures, water storage tank temperature and air temperature.

Measured signals were monitored and translated using an analog-to-digital converter and personal computer system. With the exception of the temperature measurements, all remaining measurement signals were measured in volts and converted to engineering units by the data acquisition system. The translation of the measured voltage signal to an "online" engineering unit value for the benefit of the data acquisition operator was an approximate calculation during testing. Raw voltage data were later translated into engineering units during data evaluation. This translation included compensation for initial instrument zero shift, time-dependent instrument zero shift and instrument line water column density changes (air temperature dependent).

The recorded measurements from the pressure drop tests were converted to appropriate engineering units. The pressure drop characteristics measured across the various spans in both the TFTR and CRDL were used to determine the form loss coefficients for the hardware components.

The measured pressure drop is equivalent to the sum of the friction and form loss (contraction/expansion) pressure drops.

$$\Delta P_{measured} = \Delta P_{friction} + \Delta P_{formloss}$$

The form loss pressure drop is equivalent to the following.

$$\Delta P_{formloss} = \frac{K\rho V^2}{2g}$$

Where	K	:	Form loss coefficient
	ρ	:	Water density
	V	:	Water velocity
	g	:	Gravitational constant

Therefore,

$$K = \frac{\Delta P_{measured} - \Delta P_{friction}}{\frac{\rho V^2}{2g}}$$

[

e.

]

The form loss coefficients and Reynolds Numbers determined during each steady-state condition were averaged at each condition. The resulting series of form loss coefficients and associated Reynolds Numbers were curve fitted to yield a hardware form loss coefficient as a function of Reynolds Number. The curve fits allowed the direct comparison of results from two or more pressure drop tests at the same Reynolds Number.

The table below shows the form losses at a Reynolds number of 500,000 for both the Mark-B10 and B11 assemblies.

Assembly	B10	B11
Lower End Fitting	[d.	]
End Grids(2)	[d.	]
First ISG(NMV)	[d.	]
ISG	[d.	]
Upper End Fitting	[d.	]
Total Assembly	[d.	]
Flow Area, ft^2	[d.	]

Since some of the components of the Mark-B10 and B11 designs are similar, the pressure drop and form loss values from previous tests serve as a benchmark to demonstrate similar or equivalent performance and to demonstrate reproducibility of the results.

Question 5: *In section 7.1, it is shown that in a mixed core with Mark-B11 and Mark-B10 fuel assemblies, the differences in local pressure drop at the grid spacer locations result in flow being diverted from the Mark-B11 assembly to the Mark-B10 assembly. Conversely, the lower bare-rod friction pressure drop in the Mark-B11 assembly results in a somewhat lesser flow diversion the other way, back into the Mark-B11 assembly from the Mark-B10 assembly. This flow behavior raises concerns regarding the applicability of the departure from nucleate boiling (DNB) correlation for Mark-B11 fuel mixed core configurations, because the DNB correlation is based on data that implicitly assume an essentially homogenous core. For the limiting case of one Mark-B11 assembly (which contains the hot channel) in a Mark-B10 core:*

a) What is the effect of this flow redistribution on the hot channel flow rate, in percent change in flow rate at the location of DNB, (compared to the hot channel flow rate at the same location in a full Mark-B11 core, for the same conditions of hot assembly radial peaking, system pressure, inlet temperature, total core flow, and total core power)?

b) What is the corresponding percent change in enthalpy at the location of DNB for this configuration compared to a full Mark-B11 core?

c) How does the effect of this limiting mixed core configuration on flow and enthalpy distribution vary over the full operating range of flow rate, pressure, and inlet temperature? Is it uniform over the full range of operating conditions? If is it not uniform, what are the conditions that show the largest change in hot channel flow rate, when comparing a full Mark-B11 core to a mixed core with one Mark-B11 assembly as the hot assembly?

Response 5) While it is true that most CHF correlations are based on data that implicitly assume an homogenous core, FCF has conducted tests that demonstrate the BW local condition series of correlations (BWC, BWCMV and BWU) accurately represent mixed core conditions when the thermal-hydraulic code is correctly modeled as such:

In mixed core conditions, the possibility of large axial velocity upsets at or around dissimilar grids exists. These upsets imply different local thermal-hydraulic conditions in surrounding subchannels. It has been questioned as to whether traditional steady state CHF correlations are applicable in this instance.

The FCF CHF correlation form (BWU) is composed of three parts: a uniform part dependent solely on the local thermal-hydraulic conditions of pressure, mass velocity and thermodynamic quality at the axial location of CHF, a non-uniform F factor modification dependent on the shape of the axial heat flux input, and a multiplicative geometric factor dependent on the overall fuel assembly grid spacing and heated length. It is with the uniform, local conditions part that the mixed core conditions question surfaces.

CHF correlations are developed from data from full length electrically heated bundles in 5-by-5 rod arrays. For each data point, the inlet conditions of coolant mass velocity, pressure and temperature are known, as is the power (heat flux) required to produce a DNB event. The local thermal-hydraulic conditions at the axial location of CHF must then be calculated with a computer code (for FCF, the LYNXT code).

The proof of applicability of a CHF correlation, then, is how well it can predict the critical heat flux that was measured in the DNB event using the calculated local conditions. Thus, the applicability of a CHF correlation is dependent not only on its form and data base, but on the accuracy with which the local conditions can be calculated in any given situation.

Because of the size of the test section (a 5-by-5 rod array) and the use of a series of single spacer grids (axially), normal CHF tests do not exhibit large axial offsets. FCF, however, has performed one test with widely varying subchannel axial resistances producing the large velocity upsets representative of mixed core conditions. This test was a 5-by-5 test of the Mark B zircaloy grid modeled as the corner intersection of four fuel assemblies. LDV testing of the intersection grid showed velocity upsets of as much as two to one between the intersection subchannel and the surrounding unit cell subchannels.

This test was conducted at the Babcock & Wilcox Alliance Research Center and is documented in BAW-10143P-A (BWC correlation of Critical Heat Flux, April, 1985). In the topical, the measured to predicted (M/P) CHF results were compared for two traditional test bundles and the intersection bundle.

The guide tube bundle (B15) had an average M/P of 0.971, the unit cell bundle (B16) 0.985 and the intersection bundle (B17) 0.976. The difference in M/P results is statistically insignificant. This qualified the BWC correlation for use with the Mark B fuel assembly design.

The local conditions necessary for the BWC correlation were calculated with a thermal-hydraulic computer code. The local conditions for the normal unit and guide tube bundles had very little axial upset, while the intersection bundle (which produces conditions representative of a mixed core) had severe upsets resulting from the two to one velocity upsets. The fact that the BWC correlation performed consistently on conditions representative of both homogeneous and mixed cores confirms that the FCF local conditions CHF correlations are valid for both homogeneous and mixed core applications as long as the local conditions can be accurately predicted by the subchannel thermal-hydraulic computer code.

The worst case scenario for flow re-distribution occurs when a single Mark-B11 assembly is inserted into a Mark-B10 core. Although the overall Mark-B11 pressure drop is less than the Mark-B10, the spacer grid form loss is greater due to the mixing vanes. The higher component form losses are balanced by higher friction losses in the Mark-B10. However, the friction loss occurs along the entire length of the assembly whereas the form loss occurs within the height of the grid. The form loss difference induces the highest local pressure variation and resulting cross flow into and out of adjacent assemblies. Other configurations tend to decrease the differential and therefore produce smaller cross flow effects. Comparisons of the mixed core configuration with a full core show a flow degradation of less than – [b, c] in the B11 assembly at the point of MDNBR. The associated maximum change in enthalpy was less than [b, c]. These comparisons were made for a spectrum of cases used in analyzing axial power shape limits during steady state and transient operation. The effects of varying operating conditions on mixed core change in local enthalpy and flow at the point of MDNBR are less than the effects of varying axial power shape. In all cases the MDNBR of the Mark-B11 increases due to the enhanced thermal hydraulic performance of the mixing vane grid.

Question 6: *Do the baffle cross flow conditions at the core periphery bound the cross flow conditions between mixed cores of Mark-B11 assemblies and other* *assembly designs?*

Response 6: A twelve-channel LYNXT model was used to investigate the cross flow velocity in a mixed core. The model consisted of [d] axial nodes equally spaced along the length of the fuel pin. A spectrum of cases at full flow and power conditions was run with vary axial power shapes. The highest cross flow velocity computed occurred at the top of the pin below the upper end fitting. The maximum local cross flow velocity was less than [d] ft/sec. This value is considerably less than the local cross flow velocities, [d] ft/sec, that are indicated to occur at the core periphery for similar Mark-B cores with higher pressure drop characteristics. This value is also less than the maximum span average cross flow velocity design criterion of [d] ft/sec.

Question 7: *How many Mark-BZ assemblies have been irradiated to date and what is the burnup distribution of these assemblies and the post-irradiation results to date?*

Response 7: As of October 30, 1998 a total of 2,221 Mark-B FAs with zircaloy intermediate spacer grids have been discharged. The burnup distribution is shown in Figures 7-1.

The post irradiation exams on Mark-BZ assemblies with recrystallized annealed (RXA) guide tubes of modern design are shown in Table 7-1. The results of those exams is shown in the following figures:

Figure 7-2, Fuel Assembly Growth vs Burnup, Mark-BZ Designs

Figure 7-3, Fuel Rod Shoulder Gap vs. Assembly Burnup

Figure 7-4, Max Fuel Rod Oxide Thickness vs Rod Burnup

Figure 7-5, Max Guide Tube Oxide Thickness vs. Assembly Burnup

Figure 7-6, Spacer Grid Growth vs Assembly Burnup

Question 8: *Have the cruciform leaf holddown springs in the upper end fitting been visually examined for cracking and other distortions? If so, what were the examination results along with fast fluence or burnup levels of the assemblies examined?*

Response 8: The four lead assemblies with the cruciform holddown spring (Mark-B10) were examined in detail after 3 cycles and a burnup of 46.3 GWd/mtU. Also examined was a discharge batch of Mark-B10 Fuel assemblies with a maximum assembly burnup of 51.3 GWd/mtU which is the lead use assembly with a cruciform holddown spring. In these examinations no cracking or deformations of the holddown spring, retaining nut and retainer have been observed.

Question 9: *The discussion of the design of the spacer sleeves in relation to the grid spacers is not clear.*

a) Please provide a schematic of the spacer sleeves around the guide tubes and their relation to the spacer grids along with a better explanation of how this system works to prevent grid movement.

b) The last paragraph of Section 3.1.2.3 is not clear. Please provide a further explanation on how the Mark-B11 design lowers the loads in the restraint sleeves of the upper intermediate grids (with mixing vane grids) and what design changes were made to strengthen the grid-to-sleeve interface of the Mark-B11 design. Also Section 4.2.4 states that there are positive margins in grid-to-sleeve interface loads under normal operation and faulted conditions. Please provide these margins.

Response 9: The Mark-B11 grid restraint system consists of [e.] instrument tube spacer sleeves and [e.] spacer grid insert tubes. All of these components are placed concentric to the standard Mark-B instrument tube. The arrangement of these components is shown in Figure 9-1.

[e.]

The flared ends of the spacer sleeves also interface with the spacer grid instrument tube cells, restraining the grids from axial movement. The spacer sleeve-grid interface carries the load of [e.]. The spacer sleeves in the top and bottom spans are only flared on the end that interfaces with an insert tube. The sleeve ends that interface with the inconel end grids are unflared, and are identical to the current Mark-B interface.

The [e.] located in each intermediate spacer grid serve to carry the accumulation of grid hydraulic loads transmitted up the fuel assembly by the spacer sleeves. They are [e.]. Thus, the [e.].

The Mark-B11 grid to sleeve interface is a result of recent improvements made generically to the Mark-B product line. The lower grid to sleeve interface for each intermediate spacer grid was modified to increase its load-carrying capability. [

e.

] resulting in a substantial increase in the interface strength between the grid and lower spacer sleeve. Stress margins are reported in the table below.

Unlike previous Mark-B grid restraint systems, the B11 has [

e.

].

Mark-B11 Grid/Sleeve Interface Margins (vs. ASME Code)			
	Normal Operating Conditions	Faulted Conditions	Handling (limited to [e.] lbs overload/underload)
Zircaloy Grid/ Sleeve Interface	[d.]	[d.]	[d.]
Inconel Grid/ Sleeve Interface	[d.]	[d.]	[d.]

Question 10: *Please provide the current results from post-irradiation examinations of the Mark-B11 lead test assemblies (LTAs)*

Response 10: The post-irradiation examination (PIE) inspection results for the Mark-B11 LTAs are shown along with the results of other Mark-B FA examinations in Figures 7-2 to 7-6.

Question 11: *Please provide the axial rod and assembly growth data that have been applied to the Mark-B11 design evaluations and identify the designs from which this data were taken. Also, provide justification of why these data are applicable to the Mark-B11 design. What is the margin for gap closure between fuel rod-to-upper-end-fitting at a rod-average burnup of 62 GWd/mtU (section 6.6)? Similarly, what is the margin to prevent the compression of the holddown spring to a solid height at a rod–average burnup of 62 GWd/mtU?*

Response 11: The data used to evaluate the Mark-B11 design is a subset of the data shown in Figures 7-2 and 7-3. The data is applicable because of the similarity of designs and performance. Both Mark-B (15x15) and Mark-BW (17x17) designs have similar fuel assembly growth and shoulder gap behavior as a function of burnup. Figure 11-1 plots the Mark-B and Mark-BW assembly growth together. Figure 11-2 plots the Mark-BW shoulder gap closure data. It can be observed that the behavior for both designs is similar. The features that all of these designs share are:

Mono-metallic zircaloy intermediate spacer grids
Inconel 718 end spacer grids
Fuel rods seated on bottom

Both the Mark-B11 and Mark-BW have mixing vanes on the top five intermediate spacer grids.

Based on all data collected to date, the margin for shoulder gap closure for the current Mark-B11 fuel design with a BOL gap of [d.] inches at 62 GWd/mtU is [d.] inches. The margin for compression of the hold down spring to solid height is [d.] inches

These margins are based on the following conditions:

Rod burnup - 62 GWd/mtU
Assembly burnup – [e.] GWd/mtU

Question 12: *What are the stress margins for assembly components (Sections 5.3.1 and 5.4.1.1) and buckling margins for the guide tubes (Sections 5.3.2 and 5.4.2.2) for normal operation, anticipated operational occurrences, and faulted conditions?*

Response 12: Design stress limit margins are provided in Tables 12-1 to 12-6. Stress limits are labeled according to the ASME code convention such as P_m (primary membrane), P_b (primary bending), Q (secondary stress), and F (peak stress). Margins against special stress limits are also included such as bearing or bolted joint categories. Guide tube buckling margins are based on critical buckling of the most enveloping span.

Question 13: *Please provide crushing load comparisons of the spacer grids from previous designs to those in the Mark-B11 design. Provide mixed core seismic-LOCA loading analyses for both previous Mark B designs and for the Mark-B11 design for the most limiting plant conditions.*

Response 13: A load comparison of the average dynamic crush strength for the Mark-B11 and Mark-BZ grids is provided in Table 4.2 of the topical report. The average crush strength is [d.] and [d.] lbs for the Mark-B11 and Mark-BZ grids, respectively. Also provided are comparisons of the elastic stiffness and structural damping. Note that the Mark-B11 grid impact properties are very similar to those of the Mark-BZ grid as expected. The Mark-B11 grid has a slightly higher strength and structural damping and lower stiffness, which serve to increase structural margin. The major design difference between the two grids is the addition of mixing vanes on the Mark-B11 grids and the sizing of the softstop and hardstop to accommodate the 0.416 inch-diameter fuel pin (vs 0.430 inch for the Mark-BZ). The changes have a [e.] effect on the grid structural properties as shown.

Table 5.1 of the topical report provides the most bounding faulted condition analysis results. Figure 13-1, attached, provides the different combinations of fuel that were evaluated. The fuel types included the [e,] and the latest generation of [e.]. A [e.] row core configuration was evaluated for all cases, which is considered the most enveloping based on the FCF faulted methods topical, BAW-10133P, Rev. 1.

The most limiting loads corresponded to [e.].

These values are reported in Table 5.1 of the topical report. The maximum load difference between all core design configurations was [d.]%. Maximum loads were obtained for the [e.] core and minimum loads were obtained for the [e.] core. All other core configurations were bounded by these two configurations. Maximum loads were shown to occur on [e.] as expected. In all cases, the Mark-B11 and Mark-BZ grids were shown to remain elastic for all mixed and full core conditions. Therefore, coolable geometry and control rod insertability requirements are met for the Mark-B11 and Mark-BZ fuel.

Figure 3-1: 5x5 LDV Test Results

[d.]

Figure 3-2: 5x5 LDV Test Results

[d.]

Figure 3-3: 5x5 LDV Test Results

[d.]

Figure 3-4: 5x5 LDV Test Results

[d.]

Figure 3-5: 6x6 LDV Test Results

[d.]

Figure 3-6: 6x6 LDV Test Results

[d.]

Figure 3-7: 6x6 LDV Test Results

[d.]

Figure 3-8: 6x6 LDV Test Results

[d.]

Figure 7-1
Burnup Distribution of Discharged Mark-BZ Type Fuel Assemblies

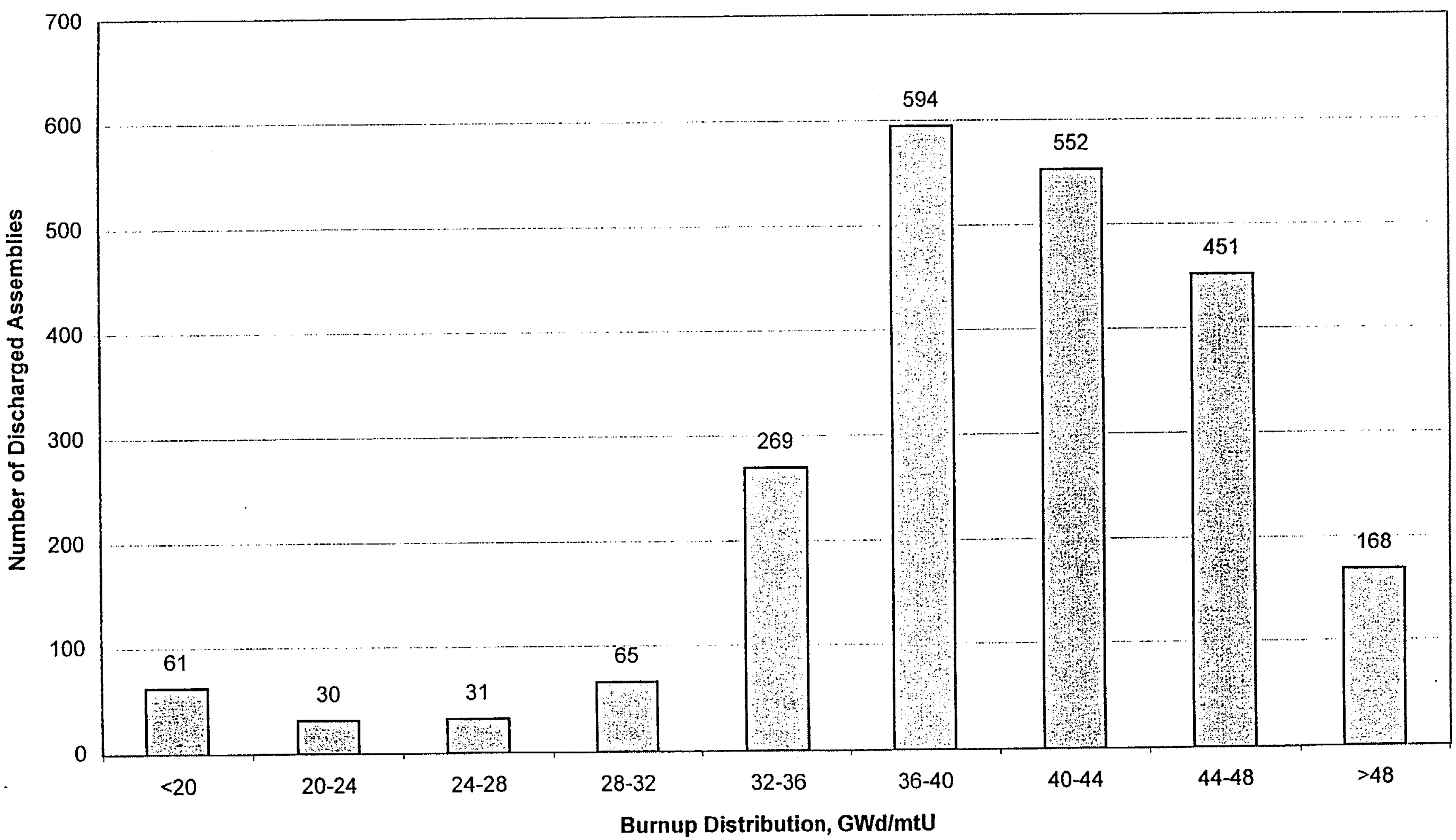

Table 7-1 PIEs on Mark-BZ FAs with RXA GTs						
FA ID	Fuel Design	Plant	Cycles Exposure	Cycles Operated	In-Core Exposure EFPD	Burnup GWd/mtU
			[d., e.]			

Figure 7-2
Fuel Assembly Growth vs Burnup, Mark-BZ Designs

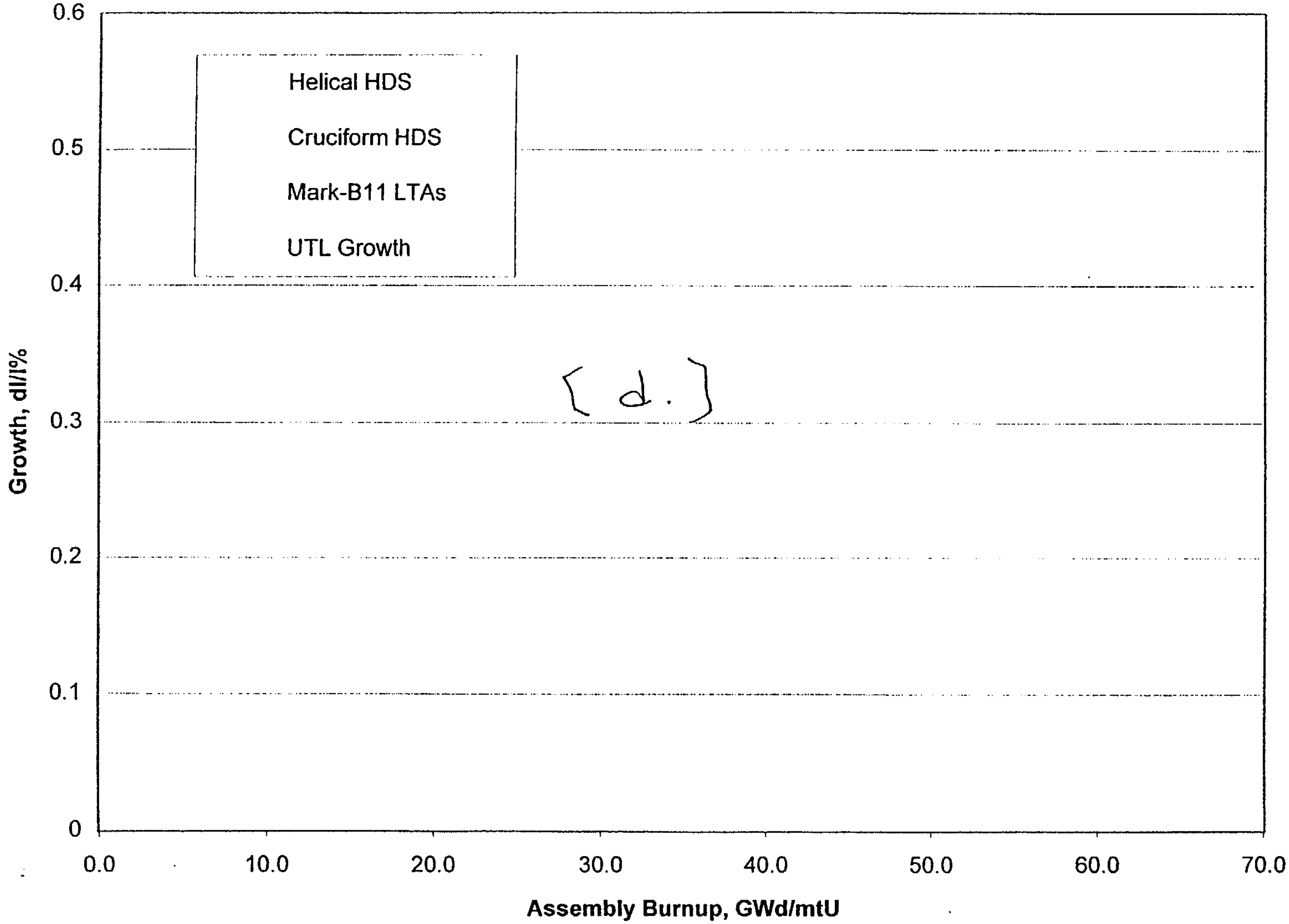

Figure 7-3
Fuel Rod Shoulder Gap vs. Assembly Burnup

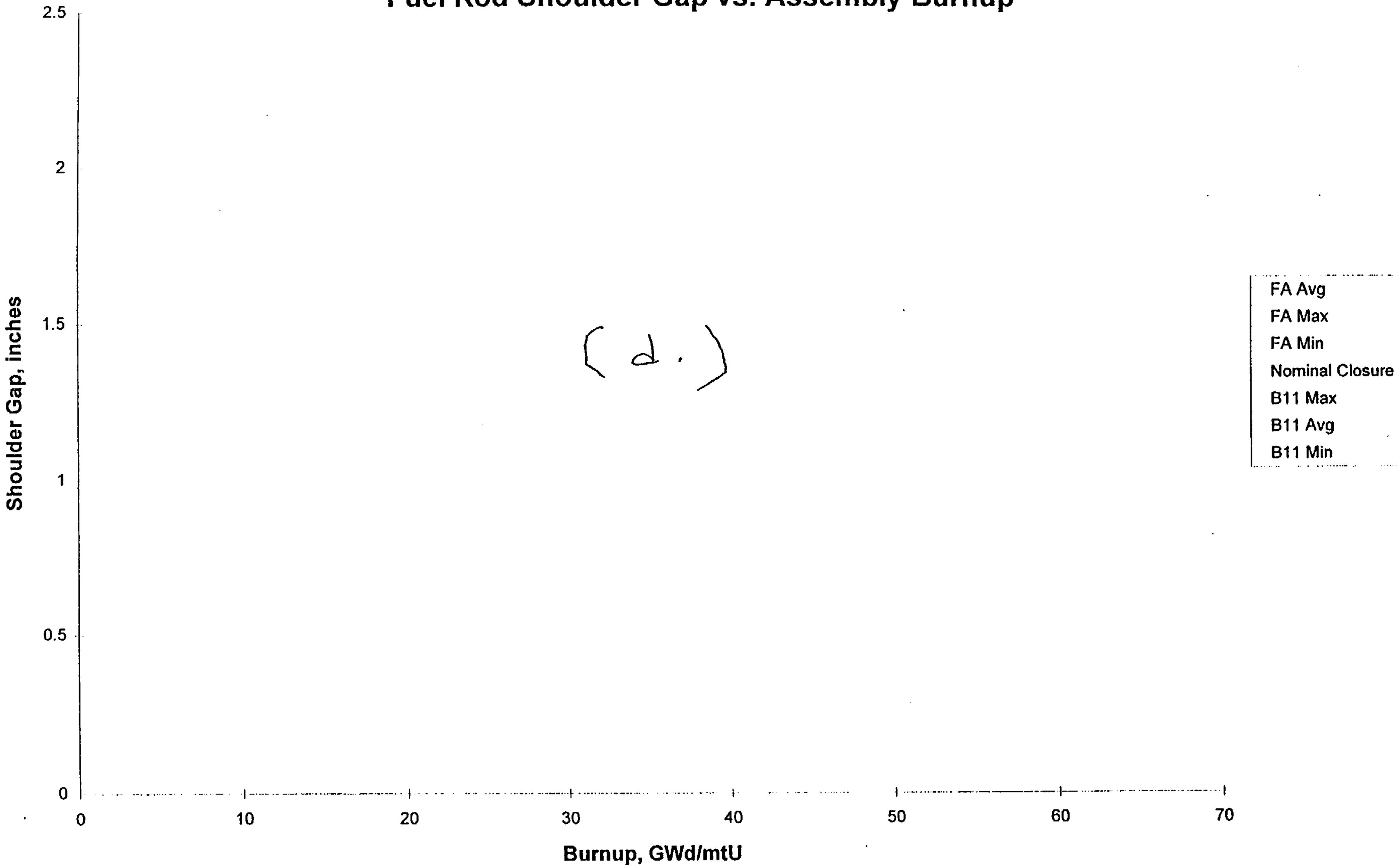

Figure 7-4
Max Fuel Rod Oxide Thickness vs Rod Burnup

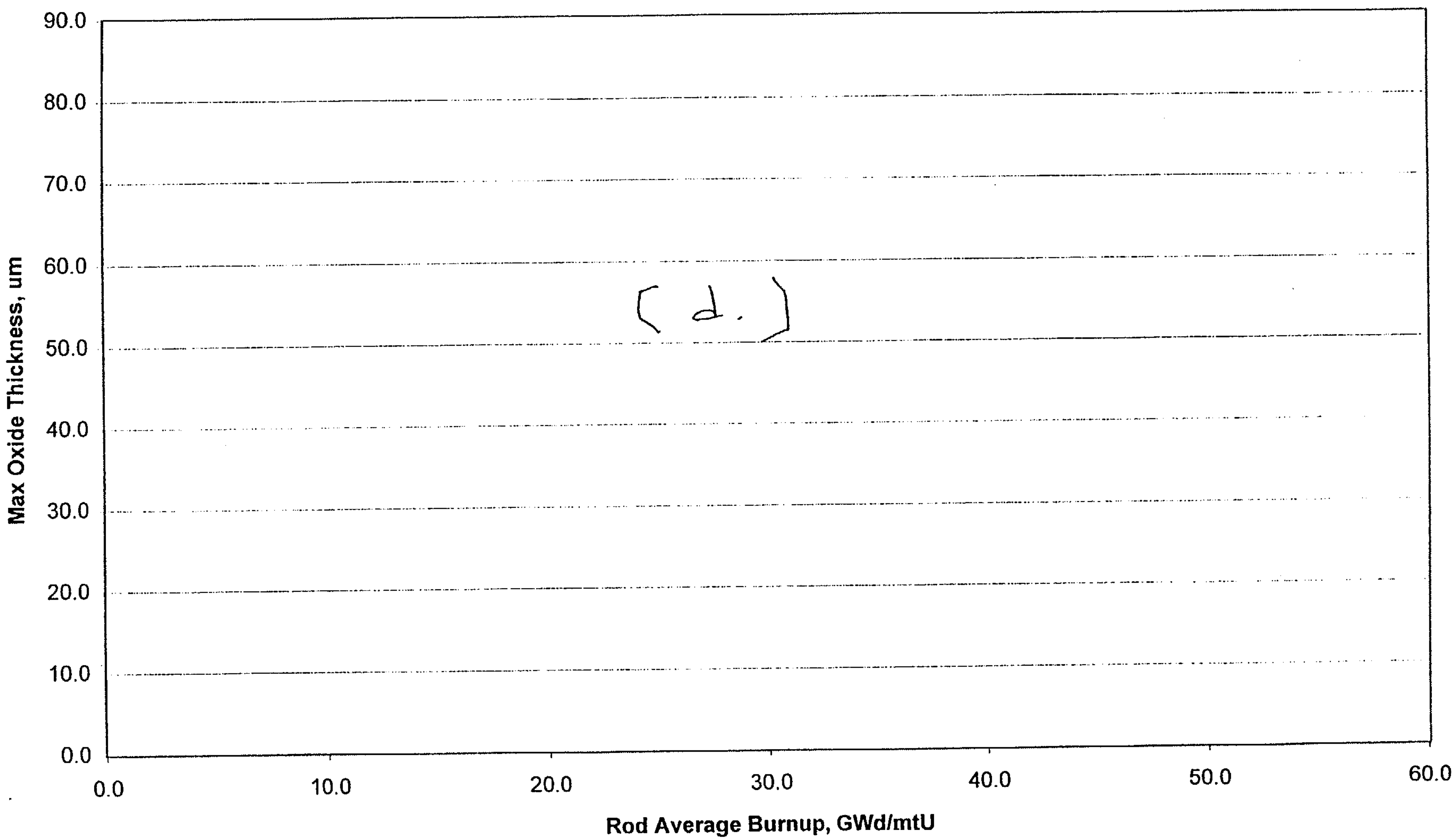

Figure 7-5
Max Guide Tube Oxide Thickness vs Assembly Burnup

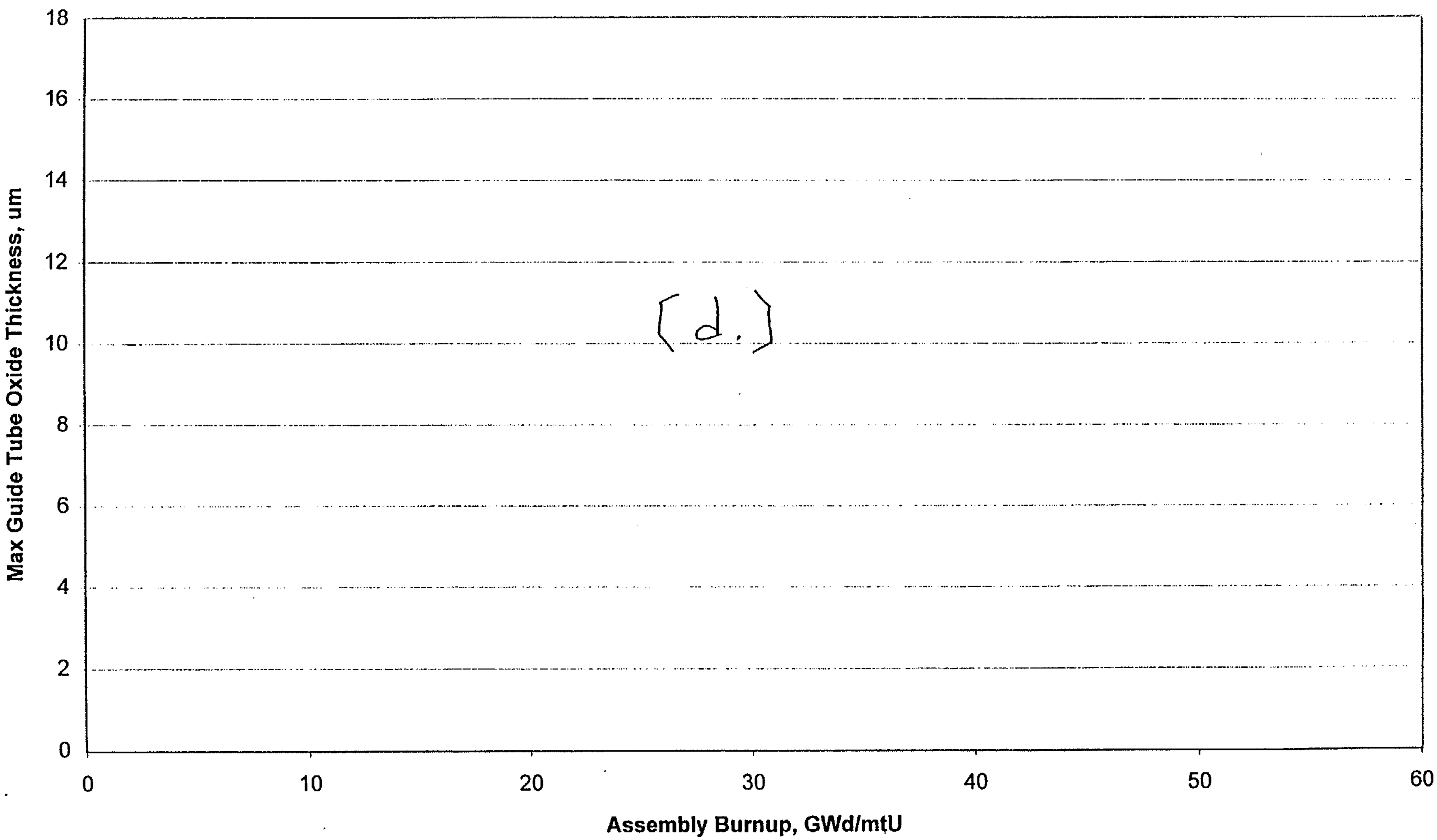

Figure 7-6
Spacer Grid Growth vs Assembly Burnup

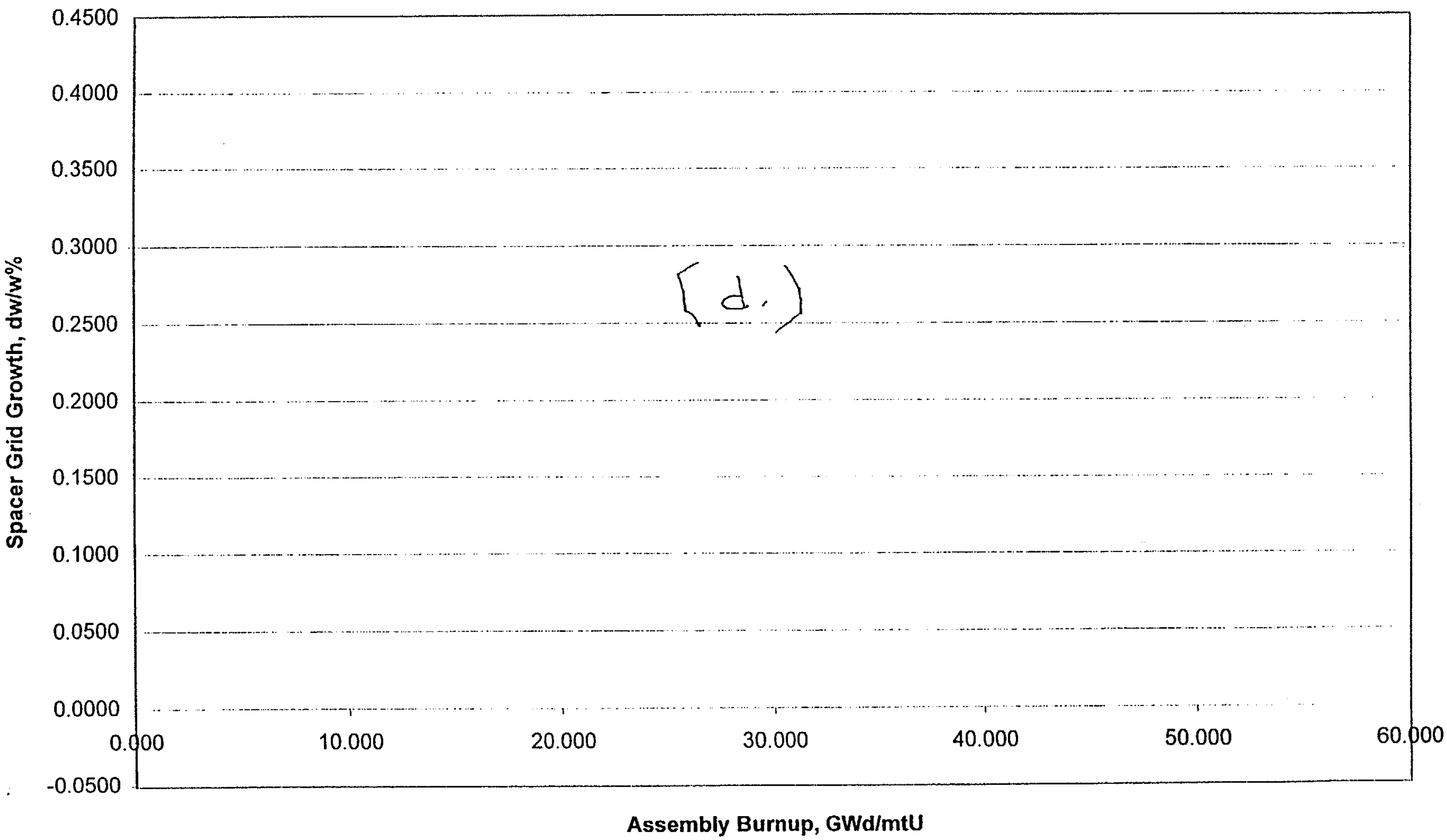

Figure 9-1: Mark-B11 Grid Restraint System

[e.]

Figure 9-2: Details of Sleeve to Grid Interface

[e.]

Figure 11-1
Fuel Assembly Growth vs Burnup

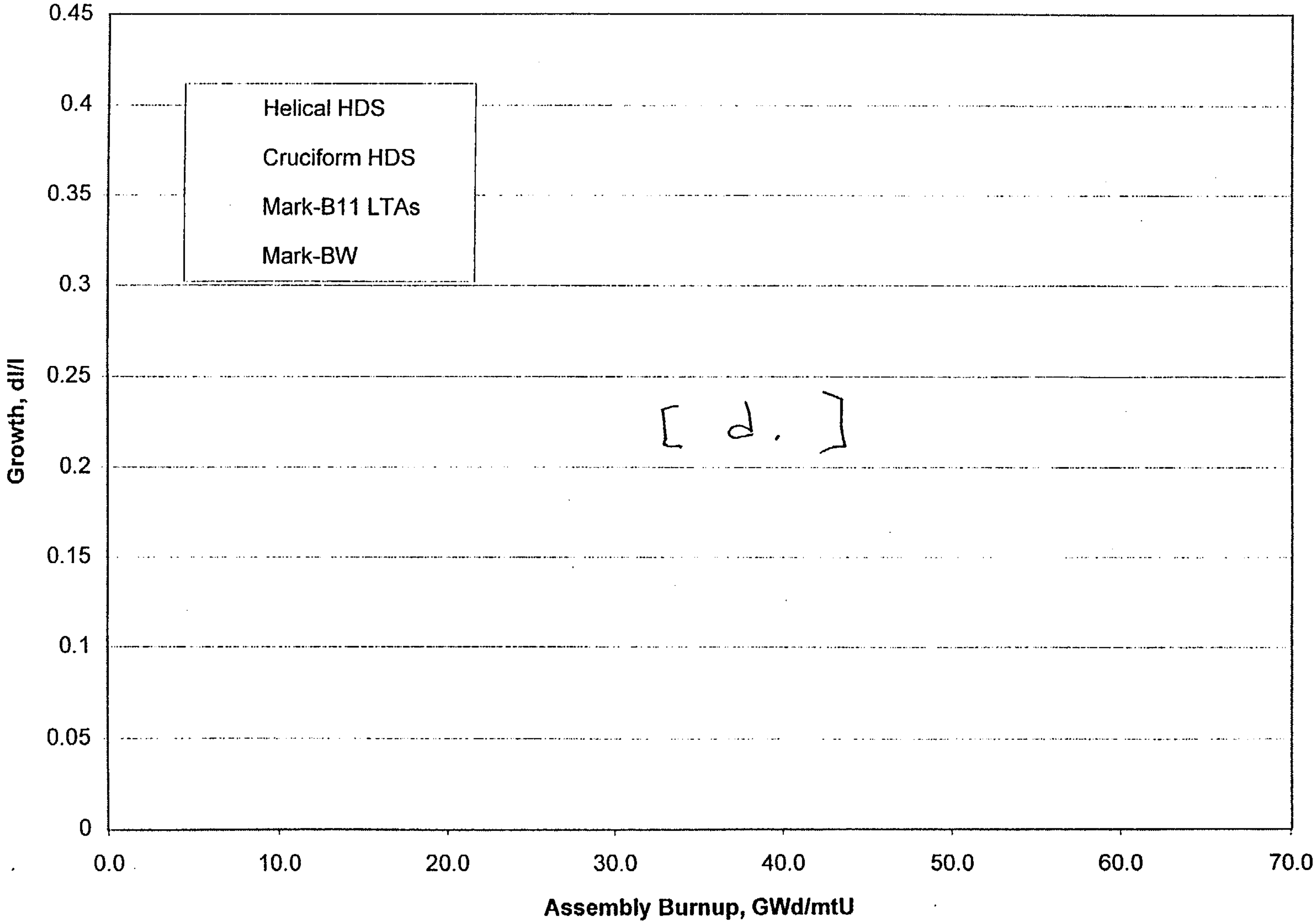

Figure 11-2
Mark-BW Shoulder Gap vs Assembly Burnup

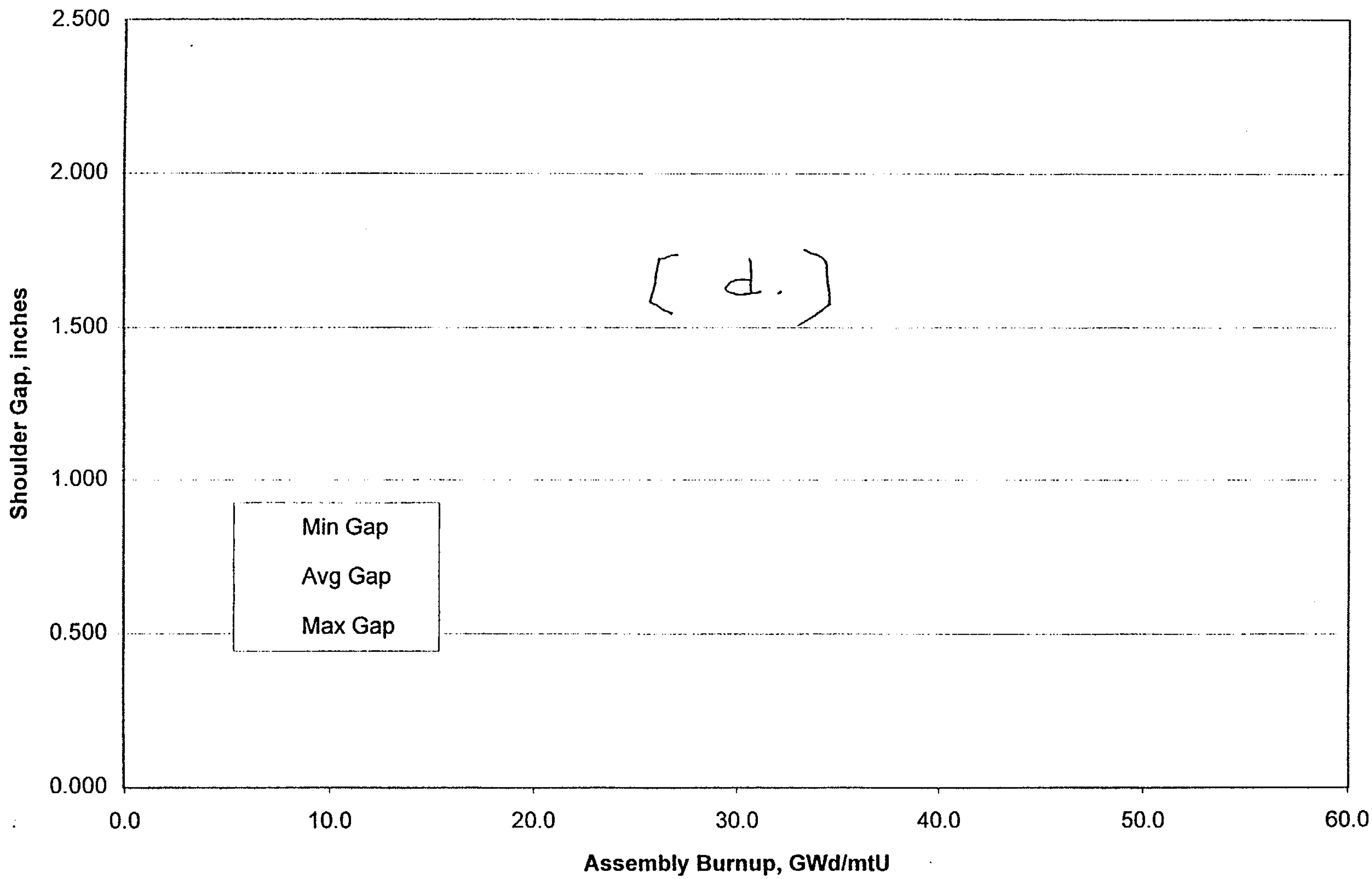

TABLE 12-1

GUIDE TUBE BUCKLING MARGINS

NORMAL OPERATION

Design criterion is [ ].

CONDITION	TEMPERATURE	% MARGIN (2)
[	e.	]
[	e.	]
[	e.	]
[	e.	]

FAULTED CONDITIONS

Design criterion is [e.].

TYPE	CONDITION	% MARGIN (2,3)
[	e.	]
[	e.	]

NOTES:

(1) For the [e.].

(2) Most limiting [c.] in the fuel assembly. Margins for other [c.] are higher.

(3) Limiting for either [c.].

TABLE 12-2

GUIDE TUBE ASSEMBLY COMPONENTS STRESS MARGINS

COMPONENT	CONDITIONS	% MARGINS			
Guide Tube Tubing	NORMAL OPERATIONS	(P_M)	(P_M + Q)		
	[	d.,e.	]		
	[	d.,e.	]		
	[	d.,e.	]		
	[	d.,e.	]		
	FAULTED(2)	(P_M)	(P_M + P_b)	(P_M + P_b) @UEF	
	[	d.,e.		]	
	[	d.,e.		]	
	[	d.,e.		]	
Lower End Plug	NORMAL OPERATIONS	(P_M)	(P_M + Q)	(P) Bearing	(P + Q) Bearing
	[		d.,e.		]
	[		d.,e.		]
	[		d.,e		]
	[		d.,e.		]
	FAULTED (2)	(P_M)	(P_M + P_b)		
	[	d.,e.	]		
	[	d.,e.	]		
	[	d.,e.	]		

NOTES:

(1) For [e.].

(2) [e.].

TABLE 12-2 (continued)

GUIDE TUBE ASSEMBLY COMPONENTS STRESS MARGINS

COMPONENT	CONDITIONS	% MARGIN				
Lock Nut Connection	NORMAL OPERATION	(P_M)	(P_M + Q)	(P_M) (3) THREAD SHEAR	(P_M+Q) (3) THREAD SHEAR	(P_M+Q) (4) BEARING
	[		d., e			]
	[		d., e.			]
	[		d., e.			]
	[		d., e.			]
	FAULTED	(P_M)		(P_M) (3) THREAD SHEAR		
	[	d., e.		]		
	[	d., e.		]		

NOTES:

(3) [e.
].

(4) [e.].

TABLE 12-3

GRID RESTRAINT SLEEVES / INSERTS STRESS (OR LOAD) MARGINS

NORMAL OPERATION

COMPONENT	CONDITIONS	% MARGINS		
Sleeve A	NORMAL CONDITIONS	(P_M)	(P_M) FLARED REGION	BUCKLING
	[	d, e		]
	[	d, e		]
Sleeve B	[	d, e		]
	[	d, e		]
Insert Tube	[	d, e		]
	[	d, e		]
Upper End Grid Sleeve	[	d, e		]
	[	d, e		]
Retainer Sleeve	NORMAL CONDITIONS	PRIMARY LOAD (WELD SHEAR)	PRIMARY LOAD (BEARING)	BUCKLING
	[	d, e		]
	[	d, e		]

FAULTED CONDITIONS

COMPONENT	TYPE	% MARGINS		
		(P_M)	(P_M) Weld Shear	BUCKLING
Sleeve A	[	d, e		]
Sleeve B	[	d, e		]
Insert Tube	[	d, e		]
Retainer Sleeve	[	d, e		]
Upper End Grid Sleeve	TYPE	(P_M)	($P_M + P_b$)	
	[	d, e	]	
	[	d, e	]	
	[	d, e	]	

NOTES:

(1) [e].

(2) [e].

TABLE 12-3 (continued)

GRID RESTRAINT SLEEVES / INSERTS
STRESS (OR LOAD) MARGINS

Mk-B11 Sleeve and Insert Tube Interfaces - % Margins		
Interface Type	Normal Operating Conditions	Faulted Conditions
Zircaloy Grid to Sleeve	[d,e	]
Inconel Grid to Sleeve	[d,e	]

TABLE 12-4

QUICK DISCONNECT MARGINS

The Quick Disconnect assembly includes the spring collar, locking springs, and guide tube collar components.

Spring Collar and Locking Springs.

Type Conditions	Component/Parameter	% Margin
Normal Operating	Collar Ear (Shear Stress)	[d]
Normal Operating	Collar Ear (Bearing Stress)	[d]
Faulted	Note (1)	[d]
Functional (Rotational Load)	Collar Weld Tab and Weld ($P_M + P_b + Q$)	[d]
Functional (Rotational Load)	Locking Spring ($P_M + P_b + Q$)	[d]
Functional (Locking/Unlocking Connection)	Locking Springs (Fatigue)	[d]
Functional (Locking/Unlocking Connection)	Collar Weld Tab and Weld (Fatigue)	[d]

Guide Tube Upper Collar.

Type Conditions	Component/Parameter	% Margin
Normal Operating (Joint Preload, 650^0F)	Guide Tube to Collar (Lip Weld Strength)	[d]
Normal Operating (Holddown Load, 70^0F)	Guide Tube to Collar (Lip Weld Strength)	[d]
Normal Operating (Holddown Load, 650^0F)	Guide Tube to Collar (Lip Weld Strength)	[d]
Normal Operating (Holddown Load, 70^0F)	Weld Prep. Area (P_M)	[d]
Normal Operating (Holddown Load, 650^0F)	Weld Prep. Area (P_M)	[d]
Normal Operating Transients	Collar Peak Stress (Fatigue)	[d]
Faulted – SSE+LOCA	Guide Tube to Collar Lip weld	[d]

NOTES

[

e.

]

TABLE 12-5

UPPER AND LOWER END FITTINGS – STRESS MARGINS

NORMAL OPERATION

COMPONENT	CONDITION	% MARGIN (Pm)	% MARGIN (Pm & Pb)
UEF	[	d, e	]
UEF	[	d, e	]
	[	d, e	]
LEF	[	d, e	]
LEF	[	d, e	]

FAULTED CONDITIONS

COMPONENT	CONDITION	% MARGIN (Pm)	% MARGIN (Pm & Pb)
UEF	[	d, e	]
UEF	[	d, e	]
	[	d, e	]
LEF	[	d, e	]
LEF	[	d, e	]

TABLE 12-6

HOLDDOWN SPRING ASSEMBLY / RETAINER STRESS MARGINS

1. Holddown [1] Spring Assembly	Condition							Fatigue [2] Usage
A. Leaf Spring	Operating Transients →							[d]
	Condition	% Margins – Threaded Fastener Evaluation Per ASME Code Guideline						
		$(P_M + Q_M)$ Shank Ave.	$(P_M + Q_M)$ Thread Ave.	$(P_M+Q_M+P_b+Q_B)$	$(P_M + Q_M)$ Head Bearing	(P_M) Max.	Shear Stress	
B. Clamp Bolt	[			d, e			]	
	Operating Transients →							[d]
C. Clamp Nut	[			d, e			]	
	Operating Transients →							[d]

2. Retainer [3]	Condition	% Margins – Stress Criteria Per ASME Code Guideline			
		P_M	$P_M + P_b$	$P_M + P_b + Q$	$P_M + P_b + Q + F$
	[		d, e		]
	[		d, e		]

NOTES:

(1) Results for most limiting/critical components of holddown spring assemblies are reported.

(2) [e].

(3) [e .]

Figure 13-1
Mark-BZ/B11 Mixed Core Configuration

[e.]