BAW-10172NP-A
Revision 0
December 1989

MARK-BW MECHANICAL DESIGN REPORT

Babcock & Wilcox

a McDermott Company

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

December 19, 1989

Mr. J. H. Taylor, Manager
Nuclear Power Division
Babcock & Wilcox
P. O. Box 10935
Lynchburg, Virginia 24506-0935

Dear Mr. Taylor:

SUBJECT: ACCEPTANCE FOR REFERENCING BABCOCK & WILCOX TOPICAL REPORT BAW-10172P, "MARK-BW MECHANICAL DESIGN REPORT" (TAC NO. 68873)

We have completed our review of the subject topical report dated July 1988 together with responses to requests for additional information dated June 2, 1989, and September 7, 1989. Based on our review we conclude that BAW-10172P provides an acceptable basis for Mark-BW fuel mechanical design. The enclosure to this letter provides our Safety Evaluation Report (SER) which details the basis and limitations of our approval. Our evaluation applies only to matters described in the topical report.

In accordance with procedures established in NUREG-0390, "Topical Reports Review Status," we request that B&W publish accepted versions of the subject topical reports, both proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions should (1) incorporate this letter and the enclosed Safety Evaluation Report between the title page and the abstract and (2) include an -A (designated acceptance) following the report identification symbol.

Should our acceptance criteria or regulations change such that our conclusions as to the acceptability of the report is no longer valid, applicants referencing this topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued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

Sincerely,

Faust Rosa for

Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation Report

SAFETY EVALUATION OF BABCOCK & WILCOX

TOPICAL REPORT BAW-10172P,

"MARK-BW MECHANICAL DESIGN REPORT"

DECEMBER 1989

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555

CONTENTS

1.0 INTRODUCTION

The Babcock & Wilcox Fuel Company (BWFC), a subsidiary of Babcock & Wilcox (B&W) has submitted to the U.S. Nuclear Regulatory Commission (NRC) a topical report, entitled "Mark-BW Mechanical Design Report," BAW-10172P (Reference 1), for review and approval. This report describes the BWFC reload licensing criteria and methods of the Mark-BW fuel design for Westinghouse-designed pressurized water reactors (PWRs).

BWFC has requested that the Mark-BW fuel assembly have a target burnup of 55 GWd/MTM for the peak assembly, 60 GWd/MTM for the peak rod, and 66 GWd/MTM for the peak pellet. BWFC has previously obtained approval (Reference 2) for application of their design criteria, analysis methods, and the Mark-B fuel assembly design, up to extended burnup levels slightly lower than those requested for the Mark-BW fuel assembly. These earlier approved design criteria generally remain applicable to the Mark-BW fuel assembly; however, some of the earlier analysis methods have been modified to be applicable to the Mark-BW design.

The TACO2 (Reference 3) and the recently approved TACO3 (References 4 and 5) fuel performance codes have been approved for extended burnup applications. The TACO3 code will be used for future safety and design analyses for the Mark-BW fuel design. The NRC staff encourages the change to the TACO3 code for extended burnup applications because it has been verified against a much larger data base at extended burnup levels than those data used to verify the TACO2 code.

In order to demonstrate the in-reactor performance of the Mark-BW design and verify the analytical models used for this design, four Mark-BW lead test assemblies (LTAs) have been loaded into the McGuire Unit 1 Reactor for irradiation during fuel cycles 5, 6, and 7. The McGuire Unit 1 Reactor and three other reactor units, McGuire Unit 2 and Catawba Units 1 and 2, are Westinghouse-designed reactors operated by Duke Power Company which will receive full batch reloads of the Mark-BW design. The core for each unit consists of 193 fuel assemblies, each with a 17x17 matrix of fuel rods.

Pacific Northwest Laboratory (PNL) has acted as a consultant to the NRC in this review. As a result of the NRC staff's and their PNL consultant's review of the topical report, a list of questions were sent by the NRC to BWFC requesting clarification of specific design criteria and licensing analyses (Reference 6). In addition, it was also requested that BWFC provide the results of additional analyses and test data that were not complete at the time of the topical report submittal. BWFC has provided responses to these questions in Reference 7.

This review was based on those licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP) (Reference 8). 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, 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 analyses. Objective 1, above, is consistent with General Design Criterion (GDC) 10 (10 CFR 50, Appendix A) (Reference 9). 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 10) for postulated accidents. "Coolability," which is sometimes termed "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 even after an accident. 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 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 anticipated operational occurrences; 2) Fuel Rod Failure Mechanisms, which apply to normal operation, anticipated operational occurrences, and postulated accidents; and 3) Fuel Coolability, which are applied to postulated accidents. Specific fuel damage or failure mechanisms are identified under each of these categories in Section 4.2 of the SRP. The BWFC design limits and analysis methods are discussed in this report under each fuel damage or failure mechanism listed in the SRP.

The purpose of the design criteria or limits are to provide limiting values that prevent fuel damage or failure with respect to each mechanism. Reviewed in this report is the applicability of the BWFC design criteria/limits to the Mark-BW design, up to the burnup levels requested. These approved design criteria/limits, along with certain definitions for fuel failure, constitute the SAFDLs required by GDC 10.

The BWFC analysis methods assure that the design limits and, thus, SAFDLs are met for a particular design application. Reviewed in this report is whether the NRC-approved design limits are met for the Mark-BW design, up to the burnup levels requested, using approved BWFC analysis methods applicable to this design.

The Mark-BW design description 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-BW design is a 17x17 Zircaloy spacer-grid fuel assembly designed for use in Westinghouse-designed reactors. The fuel assembly was designed for compatibility with the Westinghouse standard and optimized fuel assemblies. The design, however, incorporates several features that have been proven in years of in-reactor experience with the BWFC Mark-B (a 15x15 design) and Mark-C (a 17x17 design) fuel assemblies. The Mark-BW assembly description and drawings are provided in Section 3.0 of BAW-10172P (Reference 1). Therefore, BWFC has provided a satisfactory description of the Mark-BW fuel design for this review.

3.0 FUEL SYSTEM DAMAGE

The design criteria presented in this section should not be exceeded during normal operation including anticipated operational occurrences (AOOs). The evaluation portion for each damage mechanism[a] evaluates the analysis methods and analyses used by BWFC to demonstrate that the design criteria are not exceeded during normal operation including AOOs for the Mark-BW design.

(a) Stress

Bases/Criteria - In keeping with the GDC 10 SAFDLs, fuel damage criteria 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 BWFC design criteria for fuel assembly components and fuel rod cladding stresses are presented in Table 4.1 and Section 4.5.2, respectively, of the subject topical report. These criteria are based on guidelines established by Section III of the ASME Code (Reference 11). These criteria are consistent with the acceptance criteria established in Section 4.2 of the SRP (Reference 8). Consequently, these stress criteria are acceptable for application to the Mark-BW design.

Evaluation - BWFC has used the approved TACO2 fuel performance code (Reference 3) [but expects to use the recently NRC-approved (Reference 5) TACO-3 code (Reference 4) in the future] to analyze fuel rod cladding stresses for the Mark-BW fuel design, up to the burnup levels requested. The cladding stresses are calculated using the conventional thick-wall formula for a tube from engineering texts and conservative values for cladding thickness, corrosion, and temperature. BWFC has evaluated cladding stress using two cladding oxide thickness values for cladding thinning; the first is considered by BWFC to be a realistic value, while the second is considered to be bounding. Both calculated stress values are within the criteria and limits established by BWFC for the fuel rod cladding stresses; i.e., less

(a) The damage mechanisms considered are those listed in Section 4.2 of the SRP.

than the yield stress of nonirradiated Zircaloy. It should be noted that large conservatisms exist in the BWFC criterion that cladding stresses remain below the yield stress for nonirradiated Zircaloy; and this large conservatism more than compensates for the differences between the two stress analyses using the realistic and conservative oxide thickness values.

The BWFC evaluation of stresses for the other fuel assembly components have used conventional engineering formulas and ASME code guidelines (Reference 11) for analyzing primary bending and secondary stresses. From the results of these calculations in Section 4.5.2 and Table 4.1 of the topical report, BWFC has shown that the Mark-BW cladding and assembly component stresses during normal operation and AOOs are within the criteria and limits established by BWFC. Based on these acceptable ASME code guidelines we conclude that the Mark-BW fuel assembly design stresses are acceptable.

(b) Strain

Bases/Criteria - The BWFC design criteria for fuel rod cladding strain is that maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%. This criteria is intended to preclude excessive cladding deformation during normal operation and AOOs. This is the same criterion for cladding strain that is used in Section 4.2 of the SRP. The material property that could have a significant impact on the cladding strain criterion at extended burnup levels is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burnup operations' to a level that would allow cladding failure without the 1% cladding strain criterion being exceeded in the BWFC analyses.

Recent cladding ductility data from BWFC (Reference 12 and 13) and other sources (Reference 14 and 15) have shown that cladding ductility from tensile tests has decreased significantly at local burnup levels of 55 to 63 GWd/MTM (the cladding fluence ranges from 10 to 12xE21 n/cm^2). This data demonstrates that at room temperatures, the uniform cladding ductility values are much less than the 1% limit (including elastic plus plastic uniform strains). However. at the more applicable in-reactor operating temperatures of 600 to 650°F, the cladding has shown adequate (elastic plus plastic) uniform strains of 1@ or greater. This ductility decrease is significant because at burnup levels between 40 to 50 GWd/MTM (fluence range from 7xE21 to 9xE21 n/cm²). the cladding ductility values were a factor of two higher at in-reactor operating temperatures than those observed at burnups between 55 to 63 GWd/MTM. The concern here is that cladding ductility will continue to decrease with increasing burnup.

In response to an NRC question on BWFC cladding ductilities BWFC has stated that they do not believe that the earlier observations of lower cladding ductility values are representative of the cladding in their newer fuel designs. These newer BWFC designs have only minor cladding fabrication changes from the previous designs, and have been irradiated to local fluence levels between 8 to 9.5xE21 n/cm², without a significant reduction in measured cladding ductility. Therefore. BWFC has stated that they do not expect cladding ductility to be limiting at higher burnups and fluence levels in the newer fuel designs such as for the Mark-BW fuel design.

It has been shown that even with the loss-in-cladding ductility observed at local burnups up to 63 GWd/MTM (fluences to 12xE21 n/cm²). the measured uniform ductility at in-reactor operating temperatures is still within the 1% limit of combined elastic plus plastic strain imposed by Section 4.2 of the SRP. The peak pellet (local) burnup level of 66 GWd/MTM requested for the Mark-BW design is only slightly above the maximum local burnup levels of the ductility data; i.e., 63 GWd/MTM. Therefore, it is concluded that adequate ductility is expected to exist in the Mark-BW design up to the burnup levels requested by BWFC. Based on the acceptable 1@ strain limit in the SRP we conclude that the Mark-BW design strain limit of 1% is also acceptable.

Evaluation - BWFC has performed a bounding analysis using the TACO2 fuel performance code to demonstrate that the Mark-BW fuel design meets the 1% cladding strain limit for all expected operating conditions including all AOOs. Therefore, we conclude that the Mark-BW design strain is acceptable for normal operation including AOOs.

(c) Strain Fatigue

Bases/Criteria - The BWFC design criterion for cladding strain fatigue is that the cumulative fatigue usage 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 16) for fatigue usage. This criterion is consistent with SRP Section 4.2 and, therefore, we conclude that this strain fatigue criterion is also acceptable for the Mark-BW design.

Evaluation - The BWFC methodology for evaluating strain fatigue for the Mark-BW fuel design uses the O'Donnell and Langer curve for irradiated Zircaloy (Reference 16). The use of O'Donnell and Langer's curve and analysis methods for determining strain fatigue life is consistent with SRP Section 4.2 and have been previously approved by the NRC. The system transients and the number of cycles for each transient type considered in this analysis are listed in Tables 4.10 and 4.11 of the topical report (Reference 1). The BWFC methodology also requires conservative inputs of minimum as-fabricated cladding thickness, oxide layer thickness, internal fuel rod pressure, external system pressure, and differential temperature across the cladding. BWFC has evaluated cladding fatigue for the Mark-BW fuel design using two oxide thickness values for cladding thinning. The first is considered by BWFC to be a more realistic value for oxide thickness, while the second value is considered to be a bounding value for oxide thickness up to the burnup level requested in the submittal. It is acknowledged that the bounding waterside oxide thickness value used is indeed bounding, as claimed by BWFC, based on the measured oxide thickness values observed to date, but the realistic oxide value used in the first analysis may be non-conservative for particular operating plants in the U.S. However, we have concluded that there is considerable conservatism in the BWFC analysis methodology, i.e., the O'Donnell and Langer strain fatigue curve and analysis methods, for strain fatigue and this conservatism more than compensates for the differences between the realistic and bounding oxide thickness values used in these two analyses. Both analyses have resulted in fatigue usage factors that are well under the limit of 0.9 required by BWFC. Therefore, we conclude that the BWFC strain fatigue analysis methods and analyses are acceptable for the Mark-BW design.

(d) Fretting Wear

Bases/Criteria - The BWFC design criterion against fretting wear is that the fuel design shall provide sufficient support to limit fuel rod vibration and cladding fretting wear. This design criterion can also be applied to other fuel assembly components that are susceptible to fretting wear, such as the fuel assembly guide tubes. This criterion is consistent with Section 4.2 of the SRP and is found to be acceptable for the Mark-BW design.

Evaluation - BWFC has stated that they have not experienced a fretting wear problem in their previous fuel designs and, therefore, BWFC does not expect to have a fretting problem with their Mark-BW design because it is based on their previous fuel designs. In order to substantiate this, BWFC has performed ex-reactor hydraulic flow tests to simulate in-reactor vibrations on a prototype Mark-BW assembly for 500 and 1000 hours, and extrapolated the results of the ex-reactor measured wear to the projected in-reactor life of the Mark-BW assembly (Reference 1). These extrapolations of the ex-reactor measured data have shown that the expected wear is small and no fuel rod failures due to fretting wear are predicted up to the in-reactor life of this design. It should also be noted that the prototype assembly on which these tests were performed have simulated Inconel end spacer and Zircaloy intermediate spacer relaxation by intentionally oversizing the cells of the spacer grids. BWFC has also presented in-reactor fretting wear measurements from lead test assemblies with Zircaloy intermediate grids similar to those in the Mark-BW design and found no evidence of wear up to a burnup of 37.6 GWd/MTM. Based on the fact that fretting wear has not been observed in past BWFC fuel designs, and the fact that BWFC continues to perform postirradiation examination (PIE) of the Mark-BW LTAs, we conclude that the fretting wear has been adequately addressed by BWFC for the Mark-BW design.

(e) Oxidation and Crud Buildup

Bases/Criteria - Section 4.2 of the SRP identifies cladding oxidation and crud buildup as potential fuel system damage mechanisms. General mechanical properties of the cladding are not significantly impacted by thin oxides or crud buildup. The SRP does not establish specific limits on cladding oxidation and crud but does specify that their effects should be accounted for in the thermal and mechanical analyses performed for the fuel. BWFC has stated that these effects are accounted for in their Mark-BW design analyses, and therefore, this is found to be acceptable.

Evaluation - BWFC has stated (Reference 7) that the Mark-BW methodology used to account for the effect of oxidation depends upon the analysis and the version of TACO used for the analysis. Previous Mark-BW design analyses have used the TACO2 methodology to account for oxide thicknesses. The NRC staff has concluded in the safety evaluation report (SER) of the TACO2 code that the specific oxide thickness values used from this code were not conservative; however, other conservatisms in these analyses more than compensated for the lack of conservatism in the oxide thicknesses at extended burnup levels.

BWFC has stated that future oxidation analyses will be performed with the recently approved TACO3 code. The use of the oxide thickness values from Appendix I of TACO3 are found to bound the measured oxide thicknesses for past BWFC designs, and, therefore, will most likely bound those for the Mark-BW design. BWFC has indicated that they plan to perform oxide measurements of the fuel rods from the Mark-BW LTAs in order to confirm that oxide thickness values are consistent with those in Appendix I of TACO3. Therefore, we conclude that the application of TACO3 oxide cladding thickness values in the evaluation of cladding strain, fuel temperature for LOCA, and rod pressure for the Mark-BW design is acceptable.

BWFC has stated (Reference 7) that a more realistic oxide thickness value will be used for determining cladding stress and stress intensities used in the stress and fatigue usage calculations, respectively. The use of realistic oxide thickness values in the calculation of cladding stress and stress intensities for these analyses has previously been accepted by the NRC staff (Reference 2). This previous acceptance was based on the significant conservatisms in BWFCs use of the nonirradiated yield stress limit for their stress analysis, and their use of the O'Donnell and Langer analysis methods and fatigue curve (Reference 16) for their fatigue analysis.

BWFC has indicated, by a telephone conference call, that the effect of cladding crud is inherently included in their oxidation thickness values used for the thermal and mechanical analyses. The effect of crud is included because BWFC's oxidation values are based on oxide thickness measurements that include both oxidation and adherent crud in the measurement. It should also be noted that the nonadherent crud that was observed on fuel rods irradiated in the early 1970's has been virtually eliminated with todays water chemistry controls and, therefore, is not a problem with todays PWR fuels. Therefore, we conclude that the BWFC approach that inherently accounts for cladding crud in their estimate of oxide thickness is acceptable.

(f) Rod Bowing

Bases/Criteria - The BWFC design basis for rod bowing is that it shall be evaluated with respect to the mechanical and thermal/hydraulic performance of the fuel assembly and that the fuel assembly shall not exhibit excessive fuel assembly bow during its operational life.

Fuel and burnable poison rod bowing are phenomena that alter the design pitch dimension between adjacent rods. Bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than placing specific design limits on the amount of bowing that is permitted, the effects of bowing are included in the safety analysis. This is consistent with the SRP guidelines. NRC staff have previously approved this criterion for extended burnup levels (Reference 2). Therefore, we conclude that the Mark-BW rod bowing criterion is acceptable. An evaluation of BWFC methods used for predicting the degree of rod bowing in the Mark-BW fuel design at extended burnups follows.

Evaluation - The methods used to account for the effect of fuel rod bowing in BWFC assemblies have been addressed in Reference 18, which has been approved for extended burnup applications (Reference 2). BWFC has indicated (Reference 7) that they intend to apply these methods and models to the Mark-BW assemblies. In order to support the application of the BWFC Rod Bow Topical Report (Reference 18) to the Mark-BW design, BWFC has indicated that those parameters critical to rod bow such as grid spacer spring forces, grid span length, and cladding moment-of-inertia, in the Mark-BW design are similar to those in previous BWFC designs that were the basis for the Rod Bow Topical Report. In addition, significant conservatisms were included in the Rod Bow Topical Report to cover any minor variations in rod bow for their various fuel designs and none of these previous BWFC fuel designs have had any fuel performance problems that are related to rod bow. Based on the fact that previous BWFC fuel designs have not had problems related to rod bow and the fact the BWFC continues to perform PIE on the Mark-BW LTAs up to the burnups requested, it is concluded that rod bow has been adequately addressed for the Mark-BW fuel design.

(g) Axial Growth

Bases/Criteria - The BWFC design basis for axial growth is that the upper-nozzle-to-fuel-rod gap and the fuel-assembly-to-reactor-internals gap shall be designed to provide a positive clearance during the assembly lifetime. This is consistent with Section 4.2 of the SRP guidelines and, thus, is found to be acceptable for the Mark-BW fuel design.

Evaluation - The BWFC models used to predict fuel rod and assembly growth are based on axial growth data from other BWFC designs, at burnup levels up to 50 GWd/MTM. The major concern with the application of growth data from previous BWFC fuel designs to the Mark-BW design is in the onset of pellet-cladding-interaction (PCI) assisted rod axial growth. The Mark-BW fuel rod has a smaller fuel-to-cladding gap size than previous BWFC fuel rod designs and, therefore, its fuel cladding gap will close sooner in its irradiation life than the previous designs, which may accelerate rod axial growth at a lower burnup level. BWFC has anticipated this potential increase in rod axial growth for the Mark-BW design and has assumed an accelerated growth rate at a lower burnup level in the fuel rod growth model for this design. Preliminary data from the first cycle of irradiation of Mark-BW LTAs has demonstrated that both assembly and rod axial growth are well within the BWFC models predictions for this design. BWFC has stated that they plan to continue collecting this rod and assembly growth data from the Mark-BW LTAs. BWFC has also stated that additional growth data will soon be obtained from a BWFC assembly with an assembly-average burnup of 58 GWd/MTM recently discharged from Oconee 1.

In response to an NRC question on the conservatisms used in the axial growth analyses, BWFC has stated that in order to ensure that the axial upper-nozzle-to-fuel-rod gaps and the fuel-assembly-to-reactor gaps are adequate up to the burnup levels requested, they have used models that provide a 95% confidence level that 95% of the data are conservatively predicted. These analyses have also used minimum as-fabricated axial gap sizes in their calculations. In addition, BWFC will use the growth data from the Mark-BW LTAs and from other BWFC designs to confirm that their growth models for the Mark-BW design are valid and a positive clearance will remain between the top-nozzle-and-fuel-rods, and between the fuel-assembly-and-core-internals up to the burnup levels requested; i.e., 55 GWd/MTM peak assembly-average. Based on BWFCs conservative analyses and planned PIE of the Mark-BW LTAs, it is concluded that BWFC has provided adequate assurances that axial gap clearances will be maintained in the Mark-BW assemblies up to assembly-average burnups of 55 GWd/MTM.

(h) Rod Internal Pressures

Bases/Criteria - The BWFC design criterion for fuel rod internal pressures is that they will remain below nominal system pressure during normal operation including AOOs. This design criterion is consistent with the SRP guidelines and is found to be acceptable for the Mark-BW design.

Evaluation - BWFC has utilized the approved TACO2 fuel performance code in past analyses to demonstrate that the rod pressure criterion has been met; however, it is anticipated that future rod pressure analyses for the Mark-BW fuel design will use the recently approved TACO3 code (Reference 5). The TACO3 computer code has been verified against fission gas release data from fuel rods with rod-average burnup values up to 62 GWd/MTM (Reference 4). Therefore, the application of the TACO3 code to the Mark-BW fuel rod design for the calculation of rod pressures up to rod-average burnups of 60 GWd/MTM is found to be acceptable. No rod pressure analyses have been provided in the subject submittal for the Mark-BW fuel design and, therefore, a licensee using the Mark-BW design must submit a plant specific analysis.

(i) Assembly Liftoff

Bases/Criteria - The BWFC design criterion for assembly liftoff is that contact shall be maintained between the fuel assembly and the lower support plate during normal operation including AOOs except for the 120% pump overspeed transient. In addition, the fuel assembly top and bottom nozzles shall be maintained and engaged with the reactor internals for all normal operation and AOOs. The effect of the Mark-BW assembly liftoff during the overspeed transient is discussed in the evaluation below. Other than this exception, the BWFC criterion for Mark-BW assembly liftoff is consistent with the SRP guidelines and, therefore, is acceptable.

Evaluation - BWFC analyses have shown that no liftoff is expected for the Mark-BW assembly for normal operation and AOOs, except for the 120% pump overspeed transient. We have questioned BWFC (Reference 6) on what effect the Mark-BW assembly liftoff would have on core physics and thermal hydraulics for the pump overspeed transient. BWFC responded (Reference 7) that the maximum calculated Mark-BW assembly liftoff was very small and that the lower nozzle does not 1) lift off the core locator pins, 2) result in an interference fit with adjacent grids, nor 3) alter assembly coolant flow. Therefore, BWFC has concluded that the effects on core physics and thermal hydraulics are insignificant. Based on BWFC acceptable analyses, we conclude that except for the 120% pump overspeed transient the assembly remains seated on the lower core plate for normal operation and AOOs. For the 120% pump overspeed transient, the amount of assembly liftoff is so small that there are no adverse effects on the overall performance of the Mark-BW fuel assembly. Therefore, we conclude that this small amount of assembly lift off for the 120% pump overspeed transient is acceptable for the Mark-BW design.

4.0 FUEL ROD FAILURE

Fuel rod failure thresholds and analysis methods for the failure mechanisms listed in the SRP are reviewed in the following. 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 GDC 10 (Reference 9). When the thresholds are used for postulated accidents, fuel failures are permitted, but they must be accounted for in the dose calculations required by 10 CFR 100 (Reference 10). The basis or reason for establishing these failure thresholds is, thus, established by GDC 10 and Part 100. The threshold values, and the analysis methods used to assure that they are met, are reviewed below.

(a) Hydriding

Bases/Criteria - Internal hydriding as a cladding failure mechanism is precluded by controlling the level of hydrogen impurities in the fuel pellets during fabrication and is an early-in-life failure mechanism. The hydrogen level of BWFC 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 BWFC design reviews, e.g., Reference 2, have shown that this level is below the value recommended in the SRP. BWFC has stated that this same level is also used for the Mark-BW fuel design. Consequently, we conclude that the BWFC hydriding design criterion is acceptable for the Mark-BW fuel.

Evaluation - BWFC has stated that the Mark-BW fuel design is not expected to have any problems with internal hydriding. This assessment by BWFC is accepted as long as the hydrogen level in the Mark-BW fuel is controlled. Because BWFC has not had any indications of hydride failures in their fuel designs and continues to control hydrogen levels during fuel manufacture, we conclude that the Mark-BW fuel design is acceptable with respect to internal hydriding.

(b) 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 (i.e., flattening). Because of the large local strains that would result from collapse, the cladding is then assumed to fail. It is a BWFC design basis that cladding collapse is precluded during the fuel rod and burnable poison rod design lifetime. This design basis is the same as that in the SRP and, thus, is acceptable for the Mark-BW design.

Evaluation - The analytical methods and models (References 3 and 20) used by BWFC for evaluating cladding creep collapse of the Mark-BW fuel rods have previously been approved by the NRC for earlier BWFC fuel designs. Because the materials and design of the Mark-BW fuel rods are similar to earlier BWFC designs, these analytical methods and models for creep collapse are found to be applicable to the Mark-BW design. No specific creep collapse analyses have been presented for the Mark-BW design in this submittal and, therefore, a licensee using the Mark-BW design will be required to submit a plant-specific analysis.

(c) Overheating of Cladding

Bases/Criteria - The design limit for the prevention of fuel failures due to overheating is that there will be at least 95% probability, at a 95% confidence level, that departure from nucleate boiling (DNB) will not occur on a fuel rod during normal operation and AOOs. This design limit is consistent with the thermal margin criterion of the SRP guidelines and is acceptable for the Mark-BW design.

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. The BWFC DNB methodologies for the Mark-BW fuel and for specified Westinghouse fuel designs, i.e., the BWCMV correlation, have been approved (References 22 and 23). However, the application of BWFCs BWCMV correlation to mixed cores, in order to account for the differences in thermal hydraulics between Mark-BW and various Westinghouse fuel designs, has not been addressed in this submittal or in the referenced SERs (References 22 and 23). Plant specific analyses of DNB events using an approved mixed core methodology are required for licensees using the Mark-BW design in mixed core reloads.

(d) Overheating of Fuel Pellets

Bases/Criteria - To preclude overheating of fuel pellets, BWC has indicated that no fuel centerline melting is assumed for normal operation and AOOs. This design limit is the same as given in Section 4.2 of the SRP and, therefore, is acceptable for the Mark-BW fuel design.

Evaluation - The BWFC evaluation of the fuel centerline melting for the Mark-BW design has been performed with the TACO2 (Reference 3) fuel performance code; however, BWFC intends to use the recently approved TACO3 code (Reference 4) for future fuel melting analyses. Both the TACO2 and TACO3 codes have been previously approved (References 3 and 5) for fuel performance analyses up to extended burnup levels and, therefore, we conclude that they are applicable to the Mark-BW evaluation of fuel melting. Those licensees using the Mark-BW fuel design will be required to submit a plant specific analysis of fuel melting.

(e) 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 acceptable 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 are used by BWFC and are acceptable for application to the Mark-BW fuel design.

Evaluation - As noted earlier, BWFC utilizes either the TACO2 (Reference 3) or TACO3 (Reference 4) code to show that their fuel meets both the cladding strain and fuel melting criteria and that these codes are acceptable for application to the Mark-BW fuel design.

In addition, in the past, BWFC has stated (Reference 2) that they have adopted several cladding design criteria to help mitigate the effects of PCI. In this respect, the cladding is specified so that it 1) maintains sufficient ductility over its irradiated life [see Section 3.0 (b)], 2) has a higher creep rate to accommodate PCI strains, and 3) has a better inner diameter surface finish to minimize stress corrosion cracking (SCC) attack sites. BWFC has also indicated that, as a result of these criteria, the BWFC fuel designs have shown excellent performance with respect to PCI for assembly burnups up to 50 GWd/MTM, and that additional fuel examinations are to be performed on a BWFC fuel assembly with an assembly-average burnup value of 58 GWd/MTM recently discharged from Oconee-1. We, therefore, conclude that BWFC fuel has been designed to mitigate PCI failure and that periodic PIE has confirmed this.

(f) Cladding Rupture

Bases/Criteria - There are no specific design limits associated with cladding rupture other than the 10 CFR 50 Appendix K requirement that the incidence of rupture not be underestimated. A cladding rupture temperature correlation must be used in the LOCA ECCS analysis. BWFC has used a rupture temperature correlation consistent with NUREG-0630 guidance (Reference 24). We therefore conclude that BWFC has adequately addressed the bases/criteria for cladding rupture.

Evaluation - BWFC has presented their cladding deformation and rupture model for the Mark-BW design in Reference 25 and has performed a plant specific ECCS evaluation of the Mark-BW fuel design for the Catawba and McGuire plants in Reference 26. The Mark-BW cladding deformation and rupture model in these topical reports (References 25 and 26) will be evaluated by NRC in another review. Therefore, BWFC has addressed the concern of cladding rupture for the Mark-BW fuel design in References 25 and 26, and the acceptability of the

Mark-BW cladding rupture analysis will be addressed in the Safety Evaluation Report (SER) of these topical reports.

(g) <u>Fuel Rod Mechanical Fracturing</u>

<u>Bases/Criteria</u> - 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 limit proposed by BWFC 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 yield strength of the components in the Mark-BW fuel assembly design. This design limit for fuel rod mechanical fracturing is consistent with the SRP guidelines, and, therefore, is acceptable for the Mark-BW fuel design.

<u>Evaluation</u> - 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 (d) of this SER.

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 to assure that coolability is maintained are discussed for the severe damage mechanisms listed in the SRP.

(a) 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. In order to reduce the effects of cladding oxidation during a LOCA, BWFC uses a limiting criterion of 2200°F on peak cladding temperature and a limit of 17% on maximum cladding oxidation as prescribed by 10 CFR 50.46. These criteria are consistent with SRP criteria and thus are acceptable for the Mark-BW fuel design.

Evaluation - In response to an NRC question on this issue, BWFC has indicated (Reference 7) that they have presented LOCA analysis methodology and the analyses for the Mark-BW design application for the Catawba and McGuire plants in References 25 and 26, respectively. Therefore, BWFC has addressed the concern of fragmentation of embrittled cladding for the Mark-BW fuel design in References 25 and 26, and the acceptability of this analysis will be addressed in the SERs of these topical reports.

(b) 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 to provide significant pressure pulses in the primary system. To limit the effects of an RIA event, Regulatory Guide 1.77 recommends that the radially-averaged energy deposition at the hottest axial location be restricted to less than 280 cal/g. The limiting RIA event for the Mark-BW fuel design is a control rod ejection accident.

The BWFC safety criteria for the control rod ejection accident are:

1. A control rod assembly (CRA) ejection accident shall not cause further violation of the RCS integrity.

2. The maximum fuel enthalpy for the hottest fuel rod shall not exceed 280 cal/g.

These criteria are consistent with the criteria specified in Section 4.2 of the SRP and, thus, are acceptable for the Mark-BW fuel design.

Evaluation - BWFC has stated (Reference 7) that the violent expulsion of fuel from an RIA is addressed in Section 4.4.8 of Reference 27. Therefore, BWFC has addressed the concern of violent explosion of fuel for the Mark-BW fuel design in Reference 27 and the acceptability of this analysis will be addressed in the SER of this topical report.

(c) Cladding Ballooning

Bases/Criteria - Zircaloy 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. To meet the requirement of 10 CFR 50 Appendix K, the burst strain and the flow blockage resulting from cladding ballooning must be taken into account in the overall LOCA analysis. BWFC has used burst strain and flow blockage models consistent with NUREG-0630 guidance. We thus conclude that the Mark-BW fuel design has addressed the bases/criteria for cladding ballooning.

Evaluation - Fuel rod ballooning and rupture models for the Mark-BW design are provided in Section 4.3.3 of Reference 25 and the plant specific LOCA analysis for Catawba and McGuire in Reference 26. BWFC has adopted the cladding rupture and ballooning models from NUREG-0630 (Reference 24) for application to the Mark-BW design and these are discussed in References 25 and 26. The BWFC cladding ballooning model and the plant specific analyses for Catawba and McGuire will be evaluated by the NRC in a separate review of these topical reports. Therefore, BWFC has addressed the concern of cladding ballooning for the Mark-BW fuel design in References 25 and 26, and the acceptability of the BWFC cladding ballooning analysis will be addressed in the SERs of these topical reports.

(d) 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. BWFC has adopted the SRP guidelines as the design criteria. We thus conclude that the BWFC design criteria for seismic and LOCA loads are acceptable.

Evaluation - BWFC has analyzed seismic and LOCA loads on Mark-BW fuel assemblies using the approved methodology in BAW-10133P, Rev. 1 (Reference 28) for McGuire and Catawba. The results showed that the combined loads on Mark-BW fuel assemblies were small enough such that the coolable geometry is always maintained. However, the NRC staff expressed a concern about a mixed core situation of Westinghouse and BWFC fuel assemblies which may have an adverse effect because of the interaction between the two different fuel assembly designs. In response to the NRC staff concern, BWFC performed a bounding analysis of a mixed core configuration of Westinghouse and Mark-BW fuel (Reference 29). The results showed that the Mark-BW fuel had a higher grid impact load than the Westinghouse fuel for a mixed core situation; however, the Mark-BW fuel still maintained a coolable geometry. We therefore conclude that BWFC has adequately demonstrated that the Mark-BW fuel design will maintain coolable geometry for the combined seismic-and-LOCA loads in a Westinghouse designed plant.

6.0 CONCLUSIONS

We have reviewed the Mark-BW fuel design described in BAW-10172P in accordance with the SRP, Section 4.2. We conclude that the Mark-BW fuel mechanical design report is acceptable for licensing application to Westinghouse-designed reactors up to the burnup levels requested, i.e., 55 GWd/MTM peak assembly, 60 GWd/MTM peak rod, and 66 GWd/MTM peak pellet.

Those licensees that use the Mark-BW fuel design for reload applications are required to submit the following plant specific analyses: rod pressure [Section 3.0 (h)], cladding collapse [Section 4.0 (b)], DNB analysis [Section 4.0 (c)], and fuel melting [Section 4.0 (d)]. In addition, DNB analyses of mixed cores containing Mark-BW and Westinghouse fuel designs must be performed using an approved mixed core methodology.

The issues of cladding rupture, fragmentation of embrittled cladding, and cladding ballooning for the Mark-BW fuel design have been addressed by BWFC in References 25 and 26. The issue of violent expulsion of fuel has been addressed by BWFC in Reference 27. The acceptability of BWFCs analysis of these issues for the Mark-BW design will be addressed in the respective SERs of References 25, 26, and 27.

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Revision 0
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MARK-BW MECHANICAL DESIGN REPORT

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1. INTRODUCTION

The Mark-BW fuel assembly is a 17x17, Zircaloy spacer grid fuel assembly designed for use in Westinghouse-designed reactors. The fuel assembly maintains compatibility with Westinghouse reactor internals and resident fuel assemblies. The design however, incorporates features proven in B&W Fuel Company (BWFC, formerly B&W) Mark-B (a 15x15) and Mark-C (a 17x17) fuel assemblies. Design features specific to Westinghouse designed reactors have been based on designs by Nuclear Fuel Industries (NFI), a BWFC technology licensee in Japan. NFI supplies fuel to twelve Westinghouse designed reactors in Japan, and has joined with BWFC in developing the Mark-BW fuel assembly design.

To demonstrate the in-reactor performance of the Mark-BW design, four Mark-BW Lead Assemblies (LAs) have been loaded into the McGuire Unit 1 reactor for irradiation during fuel cycles 5, 6 and 7. The McGuire Unit 1 reactor and three other reactor units, McGuire Unit 2 and Catawba Units 1 and 2 are Westinghouse designed reactors operated by Duke Power Company, which will receive full batch reloads of the Mark-BW design. The core for each unit consists of 193 fuel assemblies, each with a 17x17 matrix of fuel rods.

This report describes the Mark-BW fuel assembly design, method of evaluation, and reports the results of all completed evaluations. The results demonstrate the structural integrity and the mechanical, thermal-hydraulic, and neutronic compatibility with equivalent 17x17 reload fuel for Westinghouse designed plants, and the reliability of the Mark-BW design.

The results of the evaluations reported here are based on the design of the LAs unless noted otherwise. The design of the Mark-BW fuel assembly has been changed slightly from the LAs to achieve greater burnup capabilities. The Mark-BW fuel assemblies have a target burnup of 55,000 MWd/mtU for the peak assembly, 60,000 MWd/mtU for the peak rod, and 66,000 MWd/mtU for the peak pellet. Further changes may be made to the design of the Mark-BW fuel assembly based on the results of the irradiation program for the Mark-BW

IAs, or on the addition of design improvements. These changes are being, or will be evaluated by the same or similar methods used on the IAs. The results will be reported in the reload report of the plant in which the fuel will be used.

2. SUMMARY

The Mark-BW 17x17 fuel assemblies are designed to be fully compatible with any combination of Westinghouse Standard Lattice (STD), or Westinghouse Optimized Fuel Assemblies (OFA) which comprise the balance of the core. Four Lead Assemblies (LAs) are being irradiated in the McGuire Unit 1 reactor. All four of the LAs were built to the BWFC design, however, two of the fuel assemblies were built by Nuclear Fuel Industries (NFI) a BWFC technology licensee in Japan.

Key features of the Mark-BW fuel assembly can be divided into two groups: (1) features standard on other BWFC designed fuel assemblies and (2) unique features incorporated into the Mark-BW for compatibility with a core containing Westinghouse designed fuel. Features that have been utilized on fuel assemblies for B&W designed reactor plants and the programs on which they are used include:

- Keyable spacer grids (All).
- Locking cup type removable top nozzle (Optimized Lead Test Assembly (OLTA) for the B&W 15x15 Mark-B fuel assembly).
- Zircaloy intermediate spacer grids (Mark-BZ).
- Intermediate spacer grids not attached to the guide thimbles (Mark-B, BZ,etc.).
- Annealed Zircaloy guide thimbles (Mark-B extended burnup lead test assembly Mk-BEB and OLTA Demo's).
- Skirtless Inconel upper end grid (OLTA).
- Fuel rod on lower grillage (Mark-B, BZ, C (Mark-C a B&W 17x17 fuel assembly design)).
- Top and bottom plenum springs in fuel rod assemblies (All).

Other features have also been incorporated into the Mark-BW design for use in Westinghouse reactors. To a large extent, these features were adapted from the operating experience and technology base developed by NFI in supplying reload fuel for Westinghouse-type reactors in Japan.

These features include:

-Leaf type holddown springs

-Dashpot region in guide thimbles for control rod deceleration.

-Mixing vanes on spacer grids.

-Guide thimbles used for intermediate spacer grid restraint.

With respect to the overall fuel performance, these new features represent only a minor change from the current production BWFC assemblies.

Two Hundred Ninety Six (296) assemblies of the NFI 17x17 fuel design have been irradiated in the Ohi 1 and 2 units, Takahama 3 and 4 units, and the Sendai-1 unit with outstanding performance. A peak assembly burnup of 42,100 MWd/mtU has been reached by the NFI 17x17 fuel design. The NFI fuel assembly design incorporates proven B&W design features and technology for use in Westinghouse design reactors in Japan. NFI manufactured fuel has achieved an excellent performance record. All NFI supplied fuel assemblies have been sipped for leaking fuel after each cycle of operation. Only three leaking rods have been detected, of which two were due to baffle jetting out of more than 100,000 fuel rods irradiated.

The results of the thermal-hydraulic analysis for the IAs in McGuire 1 show that the full OFA core pressure drops, lift forces, crossflows and thermal margins are only slightly affected by insertion of the IAs and that full OFA core licensing calculations bound the OFA/IA core configuration. The same is expected for any possible combination of Mark-BW fuel assemblies and Westinghouse STDs or OFAs.

The nuclear design analysis shows that the Mark-BW fuel has neutronic behavior similar to the STD assemblies, with essentially the same moderator coefficient. The differences between the Mark-BW and the OFA are slight.

The use of the Mark-BW fuel assembly in the Westinghouse designed core either alone or in conjunction with the STD and OFA 17x17 does not adversely affect plant operation, safety margins or neutronic parameters.

3. FUEL ASSEMBLY DESIGN DESCRIPTION

3.1. Fuel Assembly Design Description

The Mark-BW fuel assembly shown in Figure 3-1, is a 17x17, standard lattice, Zircaloy intermediate spacer grid fuel assembly designed for use in Westinghouse-designed reactors. The fuel assembly incorporates many standard B&W design features while maintaining compatibility with the Westinghouse reactor internals and resident fuel assemblies. The basic design parameters of the Mark-BW are compared to those of the standard and OFA Westinghouse 17x17 designs in Table 3-1. The nozzles and end spacer grid designs are based on proven NFI designs currently in operation in Westinghouse-designed reactors in Japan. The guide thimble top section and dashpot diameters, the instrument sheath diameter, and the fuel rod outside diameter are the same as the standard 17x17 Westinghouse design. The fuel rod design has been developed based on standard B&W methods applied to the Westinghouse standard outside cladding diameter. The features of the Mark-BW fuel assembly design include the Zircaloy intermediate spacer grid, the intermediate spacer grid restraint system, reconstitutable upper end fitting, and spacer springs on both ends of the fuel stack.

The fuel assembly consists of 264 fuel rods, 24 guide thimbles, and one instrument sheath in a 17x17 square array. The annealed Zircaloy-4 guide thimbles provide guidance for control rod insertion and are attached to nozzles and Inconel end spacer grids at the top and bottom of the fuel assembly to form the structural skeleton. A reduced diameter section at the bottom of the guide thimbles acts as a dashpot and decelerates the control rod assembly during trips. The annealed Zircaloy-4 instrument sheath occupies the center lattice position and provides guidance and protection for the incore instrumentation assemblies. The fuel rod and guide thimble spacing is maintained along the length of the assembly by five vaned and one vaneless Zircaloy intermediate spacer grids.

TOP NOZZLE

The top nozzle assembly is a box-like structure of welded 304 stainless steel plates as shown in Figure 3-2. The grillage of the nozzle consists of a plate with a machined hole pattern for attaching the guide thimbles and providing flow area for the reactor coolant exiting the fuel assembly. The thickness of the grillage is approximately [ ] inch thinner than the design used on the four IAs to allow for fuel rod growth at extended burnup levels. The top surface of the grillage provides the interface for fixed core components such as thimble plug assemblies and burnable poison rod assemblies.

The top plate of the top nozzle assembly provides the handling and reactor internals interface surfaces and supports four sets of three-leaf Inconel-718 holddown springs. Two guide pins on the upper core plate engage with two holes in diagonally opposing corners of the nozzle to position the assembly during operation. The holddown springs are attached to the nozzle by Inconel-600 clamp bolts. These bolts are designed with a generous shank to head radius and special head configuration to maximize the reliability of the bolts in service. The bolted attachment is protected by clamps which are bolted to the top plate independently from the spring clamp bolts. The holddown springs are machined from Inconel-718 and heat treated to maximize the in-service performance. A tang extending from the main (top) leaf of the holddown springs is captured through a slot in the top plate to preload the spring and to capture the spring parts in the unlikely event of a spring fracture.

The top nozzle assembly is removable to allow for the replacement of fuel rods in the field. A standard-type ring nut and locking cup arrangement is used to attach the nozzle to the threaded collars at the top ends of the guide thimbles as shown in Figure 3-3. The locking cup resembles a flat washer with a thin-walled circular side extending from the outside diameter. The locking cup is sandwiched in-between the ring nut and nozzle grillage and tabs on the bottom of the cup engage with slots in the nozzle grillage to prevent rotation of the cup. The walls of the locking cup are plastically deformed into grooves in the top of the ring nut to prevent rotation of the nut relative to the cup. The nuts may be removed in the field with a special tool to overcome the locking forces of the deformations in the locking cups.

BOTTOM NOZZLE

The bottom nozzle consists of a flat 304L-stainless steel plate or grillage with legs welded to each corner as shown in Figure 3-4. The bottom nozzle engages with the guide pins in diagonally opposing corners and rests directly on the lower core plate to support the weight of the fuel assembly plus the holddown spring forces. As in all BWFC fuel assembly designs, the fuel rods are seated on the grillage of the bottom nozzle.

An alternative Mark-BW bottom nozzle design is debris resistant. The circular flow holes drilled through the grillage are smaller in size and larger in number than is typical of those used on the four LAs, the standard reload Mark-BW design and on other similar PWR fuel assembly designs to date. A comparison of the flow hole patterns of the standard and debris resistant designs is given in Figure 3-5. The flow holes are sized to filter out debris that could be captured between the grillage and the lower end spacer grid. Debris captured in the spacer grid area has been known to cause damaging wear to the fuel rods. Hydraulic flow tests will be conducted on the debris resistant bottom nozzle to verify that the effects on fuel assembly pressure drop are acceptable.

The bottom nozzle is attached to the guide thimbles by a simple bolted connection illustrated in Figure 3-6. The bottom end plug of the guide thimbles is internally threaded. The bottom end plug rests on the grillage. The guide thimble bolt is inserted through a hole in the grillage and engages the threads of the bottom end plug. The bolt head is tack welded to the underside of the grillage. A stepped diameter hole through the shank and head of the bolt is provided to vent the bottom portion of the guide thimble.

END SPACER GRIDS

The end spacer grids of the Mark-BW fuel assembly are similar to the end spacer grids of the other BWFC fuel assembly designs. The grids are fabricated from [ ] inch tall Inconel-718 strips which are slotted at the top or bottom for assembly in an "egg crate" fashion. Material thicknesses of [ ] and [ ] inch are used for the inner and outer strips, respectively. The strips are TIG welded at the top and bottom of the strip intersections to form an assembly. Punched projections on the strips form stops to support the fuel rods and saddles to support the guide thimbles and instrument tube. Each fuel rod cell contains two perpendicular sets of stops and each set consists of two hardstops near the edges of the strip opposed by one softstop at the center of the strip as shown in Figure 3-7. Keying windows are provided for the insertion of rectangular keys which open the cells (softstops) for scratch free and stress free insertion of the fuel rods during fuel bundle assembly.

The top and bottom end grid restraint designs are illustrated in Figures 3-8 and 3-9, respectively. Tabs are employed at the guide thimble locations on the top of the top end grid strips and on the bottom of the bottom end grid strips for the attachment of short 304L stainless steel sleeves. These sleeves are resistance welded to the tabs during the spacer grid assembly process. The top end grid sleeves are seated against the guide tube collars to restrain the grid from upward motion and to transmit axial compression loads from the top nozzle to the fuel rods. The bottom end grid sleeves are crimped into circular grooves in the guide thimble bottom end plugs to restrain the grid in both axial directions.

The end spacer grid designs are identical to the proven Nuclear Fuel Industries, Ltd. (NFI) designs in use in Japan. NFI developed the grid designs based on BWFC spacer grid design technology obtained under a technology exchange agreement.

INTERMEDIATE SPACER GRIDS

The intermediate spacer grids of the Mark-BW fuel assembly are similar to the intermediate spacer grids of the 15x15 BWFC Mark-BZ fuel assembly design. The MK-BZ fuel assembly is a BWFC Mark-B fuel assembly with Zircaloy intermediate spacer grids (Reference 3). With the Mark-BZ spacergrid as the design basis, mixing vanes were added and the dimensions were adjusted to optimize the hydraulic and mechanical performance for Westinghouse-type design applications. A comparison between the Mark-BZ and the Mark-BW Zircaloy spacer grids is provided in Figure 3-10.

The grids are fabricated from 2.244 inch tall Zircaloy-4 strips which are slotted at the top or bottom for assembly in an "egg crate" fashion. Material thicknesses of .018 and .021 inch are used for the inner and outer strips, respectively. The strips are laser welded in an inert gas atmosphere at the top and bottom at the strip intersections to form a grid assembly. Fuel rod, guide thimble and instrument tube supports are of the standard BWFC design previously described for the end spacer grids. The proven design features of the Mark-BZ design have been incorporated into the Mark-BW design. The standard BWFC keyable features are maintained for the Zircaloy intermediate spacer grids allowing for scratch free and stress free fuel rod insertion during fuel bundle assembly. The generous leadin features and an improved corner have been verified to facilitate ease of fuel assembly handling.

The [ ] intermediate spacer grid assemblies employ flow mixing vanes on the downstream edges (top). The mixing vanes are small tabs bent approximately [ ] to the flow direction. The purpose of the mixing vanes is to improve the thermal performance of the fuel assembly by enhancing the coolant turbulence. The mixing vane design and pattern has been verified by CHF testing and is based on proven mixing vane designs of NFI in use in Japan. Mixing vanes are not used on the lowest intermediate spacer grid since the thermal enhancement is not needed in this cooler region of the fuel assembly.

As in all BWFC fuel assembly designs, the Mark-BW intermediate spacer grids are not attached to the guide thimbles, and the grids are free to follow the fuel rods early in life as they grow due to irradiation. This feature virtually eliminates axial friction forces suspected of contributing to fuel rod bow. Gross spacer grid movement is limited by stops incorporated on the instrument tube and selected guide thimbles shown in schematic form on Figure 3-11.

The Mark-BW intermediate spacer grid restraint system is identical in function to the proven Mark-BZ restraint system. The intermediate spacer grids are allowed to follow the fuel rods as they grow due to irradiation until burnup effects have significantly relaxed the Zircaloy spacer grids.

At this burnup level, the intermediate spacer grids contact rigid stops to prevent further axial movement. The stops are short sleeves or ferrules resistant welded to twelve guide thimbles and the instrument sheath above each intermediate spacer grid. The flow mixing vanes are removed from the walls of the twelve guide thimbles and instrument tube cells which interface with the ferrules. This spacer grid restraint system employs multiple guide thimbles as restraining members rather than the single instrument tube sleeve arrangement of the Mark-BZ design. This is required because of dimensional limitations associated with the relatively large diameter instrument tube and guide thimbles and increased hydraulic forces produced by the tall Zircaloy spacer grid with mixing vanes. The locations of the twelve restraining guide thimbles are shown in Figure 3-12. There is also one ferrule attached to the instrument sheath below the top end spacer grid and below each intermediate spacer grid to prevent downward motion during shipping and handling.

3.2 Fuel Rod Design Description

The Mark-BW fuel rod design consists of UO_2 pellets contained in a Zircaloy-4 seamless tube with end caps welded at each end. There is a small diametral clearance between the cladding inside diameter and the outside diameter of the fuel pellets. The tube is sealed at both ends by welding Zircaloy-4 end plugs of slightly different designs to the cladding. The top end plug has an additional hole drilled through the center by a laser beam to permit filling the interior of the fuel rod with high pressure Helium gas. A series of springs at both ends position the fuel stack within the cladding and provide protection against axial gap formation during shipping, handling and irradiation. The fuel rod is evacuated and then backfilled with helium of high purity at high pressure prior to seal welding. The high purity helium assures good heat transfer across the pellet-cladding gap. In addition the high pressure fill gas prevents creep collapse of the fuel rod during the expected incore operation of the fuel rod. A schematic of the fuel rod design is shown in Figure 3-13.

The fuel pellets are a sintered ceramic of high density UO_2. The fuel pellets are cylindrical in shape with a truncated conical dish at each end. The ends of the pellets are chamfered. The dish and chamfer on the pellet ends reduces hourglassing of the fuel pellet at power. The diameter of the fuel pellet is controlled within very tight limits. The density of the pellets used in the IAs and to be used in the reloads for the McGuire and Catawba Units is [ ] Theoretical Density (TD).

The purpose of the fuel rod spring system is to prevent axial gaps from forming in the fuel column. Thermal and irradiation induced changes in the fuel stack length can cause gaps to form in the fuel stack. Axial acceleration of the fuel stack during shipping and handling can also produce gaps in the fuel stack. Axial gaps in the fuel stack cause power peaks in adjacent fuel rods, and allow for creep ovalization into the gap.

Differences:

The fuel rod design for the Mark-BW fuel assembly differs from the Westinghouse 17 X 17 OFA fuel rod design in the following areas:

[ ]

Items A through C are shown on Table 3-1. Item D can be seen on Figure 3-13.

The Mark-BW fuel rod fabrication is initiated by [ ] the top end plug to the cladding. The fuel stack is then assembled, checked for length, and inserted into the cladding along with the top spring. Then the bottom spring is inserted and compressed using the bottom end cap and welded closed. The finished fuel rod is dried in a Nitrogen atmosphere. After drying the fuel rod is evacuated and backfilled with high purity Helium gas. The top end plug vent hole is sealed by a laser beam, and the seal weld inspected by ultrasonic methods.

3.3. Fuel Assembly Test Program

3.3.1. Introduction

A comprehensive test program was conducted to characterize and verify the performance of the Mark-BW fuel assembly design. A complete list of the tests conducted is provided in Table 3-2. Most of these tests were conducted at B&W's Alliance Research Center (ARC). Laser Doppler Velocimeter (LDV) testing of the intermediate spacer grids was conducted by NFI in Japan, and Critical Heat Flux (CHF) testing was conducted at Columbia University in New York City. Only the ARC test program is discussed in detail in this section. The LDV and CHF tests and the results of the lift and pressure drop tests are discussed in section 3.3.6 and in sections 5.2 and 5.3.

3.3.2. Prototype Fuel Assembly

A prototype fuel assembly was fabricated for use in the test program. The end of life condition was simulated by intentionally oversizing the fuel rod cells of the spacer grids. The Inconel end spacer grids were relaxed approximately 60% and the Zircaloy intermediate spacer grids were relaxed approximately 90%. End of life conditions are the worst case for testing fuel assembly vibration and wear characteristics and mechanical properties. The fuel rod internals were typical of production parts with pressed tungsten-nickel-copper powder pellets used to simulate the fuel pellets. The fuel rods were pressurized to [ ] psig with helium to simulate actual fuel rod internal pressure. The prototype spacer grid restraint system employed four ferrules resistance welded to guide thimbles above each intermediate grid and one ferrule below each grid attached to the instrument tube with a cylindrical expansion joint. The nozzles were typical of the standard nozzles.

3.3.3. Hydraulic Flow Testing

Hydraulic flow testing of the prototype fuel assembly was conducted in the Control Rod Drive Line (CRDL) facility at the ARC. A schematic of the CRDL facility is given in Figure 3-14. The autoclave houses one fuel assembly, and one Reactor Cluster Control Assembly (RCCA) with a drive mechanism if needed for testing. The CRDL internals provide support and coolant flow geometries at each end of the fuel assembly representative of the reactor. The system is capable of producing flow rates, temperatures, pressures, and water chemistries typical of those found in pressurized water reactors. The hydraulic flow tests included lift, pressure drop, life and wear or environmental tests, and RCCA trip tests.

The prototype fuel assembly was modified by removing the holddown springs and replacing the upper ring nuts with cap nuts to block flow through the guide thimbles during the lift and pressure drop tests. Pressure drop measurements across the fuel assembly and various fuel assembly components (spacer grids, nozzles) as illustrated in Figure 3-15 were made at various flow rates at 120°F, 200°F, 300°F, 400°F, 500°F, 585°F, and 630°F. At each test temperature, the flow rate which lifted the fuel assembly off the lower support was measured. The pressure drop measurements were used to calculate form loss coefficients, and the lift test served as a benchmark for the calculations. The lift and pressure drop test results are discussed in more detail in section 5.0.

The prototype fuel assembly was returned to the standard configuration and fitted with a thimble plug assembly for the life and wear tests. Two 500 hour tests were conducted with the prototype fuel assembly subjected to simulated reactor conditions. Pressure drop measurements were repeated at the beginning of each test. Coolant temperature, pressure, flow rates, and chemistry were representative of PWR conditions. The purpose of the life and wear tests was to verify the corrosion and wear resistance of the design. In particular, fuel rod wear due to the relaxed spacer grid interface was characterized.

A thorough examination of the fuel assembly after 1000 hours of exposure to the simulated PWR environment revealed no abnormal corrosion or wear. Wear depth measurements were taken at the spacer grid contact site of selected fuel rods extracted from the fuel assembly. Fifteen fuel rods were removed from the fuel assembly for inspection after the first 500 hours and were replaced with new fuel rods. Eighteen fuel rods were removed for inspection after the second 500 hours, including three of the replacement fuel rods inserted after the first 500 hour test. The locations of the fuel rods extracted from the fuel assembly and inspected are shown in Figure 3-16. The fuel rods were rotated 45° before being withdrawn from the fuel assembly (replacement rods were rotated 45° after insertion) to prevent the contact sites from being scratched by the grids.

An evaluation of the wear depth measurements has shown that the wear pattern indicates an initial breaking-in period followed by very small long term progressive wear. After 500 hours of test, 427 of 864 contact sites (49%) exhibited measurable wear; while after 1000 hours, 570 out of 720 contact sites (79%) exhibited measurable wear with no apparent change in wear depth. Log normal statistical techniques were used to characterize the magnitude of wear at the contact sites. A comparison between the corresponding values for the 500 and 1000 hour testing periods has been used to assess the contribution of the long term progressive wear rate on the total observed wear. The mean values of the log normal distributions for the test periods indicate no significant differences between the data at 500 and 1000 hours, as shown in Table 3-3. These results show that the contribution of the long term progressive wear rate on the total observed wear is small, and no fuel rod failures due to fretting wear are predicted.

The prototype fuel assembly was also tested with a simulated reactor cluster control assembly (RCCA) in the CRDL facility. The CRDL facility was reconfigured to include simulated reactor upper internals at a RCCA core location. A Babcock & Wilcox type control rod drive mechanism was modified to eliminate the snubbing action for use in these tests. Pressure drop testing identical to the pressure drop testing previously described was conducted with the RCCA both fully inserted and fully withdrawn from the fuel assembly.

RCCA trip testing was conducted at selected flow rates at 190°F, 250°F, 550°F, and 600°F. The maximum trip time measured from RCCA movement to dashpot entry was [ ] seconds and occurred at [ ] gpm at [ ]°F. This maximum trip time is well below the [ ] seconds acceptance limit. Trip time criteria are listed in more detail in section 4.1.1.9. Trip times were measured at other combinations of flow rate and temperature to fully characterize the trip performance of the fuel assembly. A typical position versus time trace is provided in Figure 3-17.

3.3.4. Fuel Assembly Mechanical Tests

Fuel assembly mechanical tests were conducted to determine the mechanical characteristics of the fuel assembly in air at room temperature. The results from these tests were used to benchmark the analytical models described in section 4.1. The prototype fuel assembly was supported by mock core plates with guide pins to simulate the end conditions in the reactor. The frequency and damping values and the mode shapes for the first five modes of vibration of the fuel assembly were determined by a shaker test. An electrodynamic shaker was used to excite the fuel assembly near the midplane at various frequencies and amplitudes as the responses of the fuel assembly nozzles and spacer grids were measured and recorded. These displacement-time histories were analyzed to determine the dynamic characteristics of the fuel assembly. The results from these vibration tests are summarized in Table 3-4. The first five mode shapes of the fuel assembly are illustrated in Figure 3-18.

A fuel assembly pluck test was performed on the prototype fuel assembly in air and in water at room temperature to obtain fundamental natural frequency and damping values at various amplitudes. The pluck test is conducted by measuring and recording the displacement of selected spacer grids as the fuel assembly is deflected laterally at the midplane and quickly released. The results were consistent with the shaker test results for the first mode. Submersion of the fuel assembly in water reduced the fundamental frequency [ ] but had no measurable effect on the damping. This test was also preformed in air on two of the lead fuel assemblies for McGuire Unit 1 for comparison with the prototype test results. As expected, the lead assemblies exhibited a slightly [ ] natural frequency and approximately [ ] of the damping of the prototype fuel assembly with relaxed spacer grids. The results from these vibration tests are summarized in Table 3-5.

Force versus deflection tests were conducted to determine the axial, lateral, cantilever, and torsional stiffness of the fuel assembly. The fuel assembly is compressed along its longitudinal axis by an application of forces at the nozzles in the axial stiffness test. The lateral stiffness test consists of loading the fuel assembly laterally at the two center spacer grids. The top nozzle is deflected laterally while the bottom nozzle is restrained during the cantilever stiffness test. The torsional stiffness test is the application of a torque to the top nozzle such that the fuel assembly is twisted about its longitudinal axis with the bottom nozzle restrained. The results from these stiffness tests are used to benchmark analytical models and are provided in Table 3-6.

3.3.5. Fuel Assembly Component Tests

3.3.5.1. Spacer Grid Impact Tests

Impact tests were conducted to determine the dynamic characteristics of the Zircaloy intermediate spacer grids. These characteristics were used as input properties for the analytical models of the fuel assembly, to establish allowable impact loads, and to characterize the plastic deformation of the spacer grids. The Mark-BW Zircaloy intermediate spacer grids were impact tested using the standard B&W spacer grid impact test method.

The spacer grid was supported at eight diagonal guide thimble locations on a rolling cart. Fuel rod segments loaded with brass pellets, and guide thimble segments were installed into each test grid for mass and stiffness modeling. The cart and test grid were propelled into a stationary instrumented impact surface by a pneumatic ram as illustrated in Figure 3-19. Impact force, impact duration, pre-impact and post-impact velocity, and grid permanent deformation were measured during each impact. These data were input into a single degree of freedom model to calculate equivalent dynamic stiffness and damping values. Spacer grids were tested at room temperature and at 600°F. The results from the spacer grid impact test are provided in Table 3-7.

The failure mode of the spacer grids was racking in the outer three rows. Grid plastic deformation was correlated with the first plastic impact load for use in the LOCA and seismic analyses described in section 4.1.2.

3.3.5.2. Spacer Grid Static Crush

Zircaloy intermediate spacer grids were also crush tested at room temperature to determine the static stiffness and elastic load limit for use in establishing allowable grid clamping loads during shipping. The crush test was also used to determine the value of grid deformation at which localized distortion of an individual guide thimble affected insertion of a control rod. Each test grid contained a full complement of fuel rod segments loaded with brass slugs and guide thimble segments. Each grid was crushed in its plane under slowly applied forces. The failure mode of the spacer grid was racking. Closure of a guide thimble occurred after the entire grid was racked into a diamond shape.

The results of the static crush test are given in Table 3-8.

3.3.5.3. Spacer Grid Slip Tests

The forces required to slip spacer grids relative to the fuel rods, guide thimbles, and instrument tube in air at room temperature were measured during the slip tests. Inconel end spacer grids simulating the Beginning of Life (BOL) and End of Life (EOL) conditions, and Zircaloy intermediate spacer grids simulating the EOL condition were tested. The EOL Inconel spacer grid cells were intentionally oversized to reduce the interference with the fuel pins by approximately [ ]. No EOL Zircaloy spacer grids were tested because it was assumed that the minimum slip force for Zircaloy grids is [ ]. The results from this test represent the total friction force between the spacer grids and the fuel rods, and were used as inputs into analytical models of the fuel assembly. The test results are provided in Table 3-5.

3.3.5.4. Holddown Spring Force/Deflection Tests

Holddown spring force/deflection tests were conducted in air at room temperature to characterize the force/deflection behavior of the springs. The force/deflection characteristics were used in the normal operating analysis to determine fuel assembly holddown forces. The holddown spring rate calculated from the test results was used to model the holddown springs in the analytical model of the fuel assembly. The holddown springs were strain gaged to determine the surface strains as a function of force. These data were used directly in the holddown spring stress analysis. A typical holddown spring force versus deflection curve is shown in Figure 3-21.

3.3.5.5. Guide Thimble Buckling Tests

Guide thimble segments, each representing a particular span between spacer grids, were tested individually in axial compression to determine their buckling characteristics. Four segments representing the span between the upper end spacer grid and the upper intermediate spacer grid, and four segments representing the span between the two uppermost intermediate spacer grids were tested at room temperature. These two spans represent the two spans most highly loaded in compression during fuel assembly operation.

Each test segment was fabricated with hemispherical end caps on each end to allow unrestricted rotation of the ends. The specimens were loaded through special plates supported by ball bearings to allow unrestricted translation of the ends. End rotational restraint was provided by prototype spacer grid sections in the relaxed condition. The test was conducted by slowly increasing an axial compressive load while measuring and recording deflections in both the axial and lateral directions. The lateral deflection was measured at midspan of the segments and was the primary indicator of the buckling performance. The axial deflection was used to calculated axial stiffness values for guide thimble span elements in the fuel assembly analytical models.

The guide thimble segments failed as columns with initial eccentricity rather than as straight columns described in classical Euler theory. Secant formula column theory has been shown to accurately describe the buckling behavior of the guide thimble spans. Two forms of the secant formulas are given next, one in terms of the lateral displacement and one in terms of compressive stress.

3.3.5.6. Joint Strength Tests

Various structural joints were tested for stiffness and strength in air at 650°F. These joints include guide thimble end plug welds, sleeve to end spacer grid resistance welds, the bottom end spacer grid sleeve crimp connection, ferrule to guide thimble resistance welds, and the ferrule to Zircaloy intermediate spacer grid interface. The stiffnesses of these attachments were used as inputs to the analytical models of the fuel assembly, and the strengths were used to establish allowable loads in the structural evaluations.

3.3.6.1. Hydraulic Lift/Pressure Drop Test

Hydraulic lift and pressure drop testing, performed in the Control Rod Drive Line (CRDL) at Babcock and Wilcox's Alliance Research Center (ARC) in the fall of 1985, provided quantitative results which allowed for the determination of the formloss coefficients of various fuel assembly components on the Mark BW fuel assembly. The CRDL is a closed loop test facility that can provide coolant flow rates to approximately 2500 gpm, coolant temperatures to 630°F and operating pressures to 2500 psi. For the Mark BW hydraulic lift and pressure drop testing, coolant flow rate varied from 450 to 2500 gpm, coolant temperature varied from 80 to 630°F, and operating pressure varied from 680 to 2280 psia. This range of coolant conditions provided test data for Reynolds Number ranging from 50,000 to 500,000.

Testing was performed using a full scale assembly with axial pressure taps placed at fourteen different axial locations. The test data were recorded in the form of pressure differences measured, in inches of water, across seventeen different spans along the length of the assembly. The CRDL test measurement devices were calibrated to read zero inches of water with the pumps off, thereby eliminating the contribution of pressure drop due to elevation from the measured pressure drop. During the course of a test run, the data logging device in the CRDL facility reads several pressure drops from various components and then reads the coolant pressure, temperature and flowrate. This way, data for several sets of coolant conditions are available for a given test run. Pressure drop cases were performed at a variety of flow rates, for a given set of coolant temperature and pressure conditions. Hydraulic lift cases were performed such that, for a given set of coolant temperature and pressure conditions, flow rate was increased to the point at which the fuel assembly lifted. Pressure drop measurements were then made and the flow rate was reduced until the fuel assembly reseated and pressure drop measurements were made again. The results from these tests were analyzed and formloss coefficients for various fuel assembly components determined. For more information on the pressure drop and hydraulic lift characteristics of the Mark BW fuel assembly see Section 5.2 and 5.3 of this report.

3.3.6.2. LDV Test

Extensive Laser Doppler Velocimeter (LDV) testing performed by Nuclear Fuel Industries (NFI) has provided a detailed description of the subchannel flow distribution within the Mark BW fuel assembly. The results from these tests have been used to confirm the subchannel formloss coefficients which were determined analytically. The test apparatus consisted of a water flow loop, the test containment and the test rod bundle. The test rod bundle was made up of four 4X4 mini-bundles which simulated the corner regions of four adjacent Mark BW assemblies. The mini-bundles were approximately 65 inches tall and contained four spacer grids each. The middle two grids were separated by a distance equal to the distance between spacer grids on a Mark BW assembly. In order to characterize the coolant flow between the two middle spacer grids, velocity profile measurements were made at four different cross-sectional planes. These velocity measurements were measured along lines through the centers of the subchannels at planes which were 20, 100, and 200 millimeters downstream of the lower spacer grid and 20 millimeters upstream from the upper grid. All velocity measurements were normalized to an average velocity of 4.7 m/s (15.4 ft/sec). The tests were run with a water temperature between 20°C and 40°C (68°F-104°F) at a prescribed flow rate. Results from two LDV tests, one with four adjacent inconel grids and one with four adjacent zircaloy grids, have been analyzed. The results of the analysis show that the analytical results can be correlated to the test results. LDV test results for adjacent inconel and zircaloy grids are still undergoing evaluation.

3.3.6.3. CHF Testing

In January 1987, a Critical Heat Flux (CHF) test was conducted at the Columbia University Heat Transfer Research Facility to confirm the applicability of the BWCMV correlation for predicting CHF in rod bundles with zircaloy mixing vane grids like those used in the Mark BW fuel assembly. The BWCMV correlation was originally developed using data which had been compiled for test bundles having inconel mixing vane spacer grids. The new test was conducted on a full length, 5X5 test bundle with non-uniform axial heat generation. A 99 point test matrix was established which covered the entire range of the BWCMV data base. Results from the confirmatory test showed that the BWCMV correlation conservatively predicts the CHF performance of the Mark BW fuel design. For a complete discussion of the test and results see Addendum B of the BWCMV topical report (reference 22).

Table 3-1. Comparison of Mark-BW, W Std. and OFA Design

Parameter	17x17 B&W Mark-BW Fuel Assembly Design	17x17 W First Cycle Standard Fuel Assembly Design	17x17 W Reload Cycle OFA Fuel Assembly Design
Fuel Assembly Length, in.	159.8	159.8	159.8
Fuel Rod Length, in.	151.500	151.630	151.630
Assembly Envelope, in.	[ ]	[ ]	[ ]
Compatible with Core Internals	Yes	Yes	Yes
Fuel Rod Pitch, in.	.496	.496	.496
Number of Fuel Rods/Ass'y.	264	264	264
Number of Guide Thimbles/Ass'y.	24	24	24
Number of Instrumentation Tube/Ass'y.	1	1	1
Compatible with Movable In-Core Detector System	Yes	Yes	Yes
Fuel Tube Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Fuel Rod Clad O.D., in.	.374	.374	.360
Fuel Rod Clad Thickness, in.	0.024	0.0225	0.0225
Fuel/Clad Gap, mil.	6.5	6.5	6.2
Fuel Pellet Diameter, in.	0.3195	0.3225	0.3088
Fuel Pellet Density, % TD	[ ]	[ ]	[ ]
Guide Thimble Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Inner Diameter of Guide Thimbles (upper part), in.	0.450	0.450	0.442
Outer Diameter of Guide Thimbles (upper part), in.	0.482	0.482	0.474
Inner Diameter of Guide Thimbles (lower part), in.	0.397	0.397	0.397
Outer Diameter of Guide Thimbles (lower part), in.	0.429	0.429	0.429
Inner Diameter of Instrument Guide Thimbles	0.450	0.450	0.450

Table 3-1. Comparison of Mark-BW, W Std. and OFA Design (Cont'd)

Parameter	17x17 B&W Mark-BW Fuel Assembly Design	17x17 W First Cycle Standard Fuel Assembly Design	17x17 W Reload Cycle OFA Fuel Assembly Design
Outer Diameter of Instrument Guide Thimbles, in.	0.482	0.482	0.474
Composition of Grids	2 Inc.-718 End Grids, 6 Zircaloy-4	Inc.-718	2 Inc.-718 End Grids, 6 Zircaloy-4
Top Nozzle Holddown Springs	3-Leaf	3-Leaf	3-Leaf

Table 3-2

Mark-BW Prototype Test Program

Hydraulic Flow Testing

Fuel Assembly Lift

Pressure Drop with Thimble Plug

1'st 500 Hour Life and Wear

Pressure Drop with Thimble Plug

2'nd 500 Hour Life and Wear

Spacer Grid LDV

Critical Heat Flux (CHF)

RCCA Drag

Pressure Drop with RCC Assembly

RCCA Trip

Fuel Assembly Structural

Frequency and Damping in Air

Frequency and Damping in Water

Lead Assembly Frequency and Damping in Air

Axial, Lateral, Torsional, and Cantilever Stiffness

Component Structural

Spacer Grid Impact

Spacer Grid Crush

Spacer Grid Slip

Holddown Spring Force/Deflection

Guide Thimble Buckling

Joint Strength

Control Rod Insertion

Table 3-3

Wear Measurement, All Contact Sites

Table 3-6

Fuel Assembly Stiffness

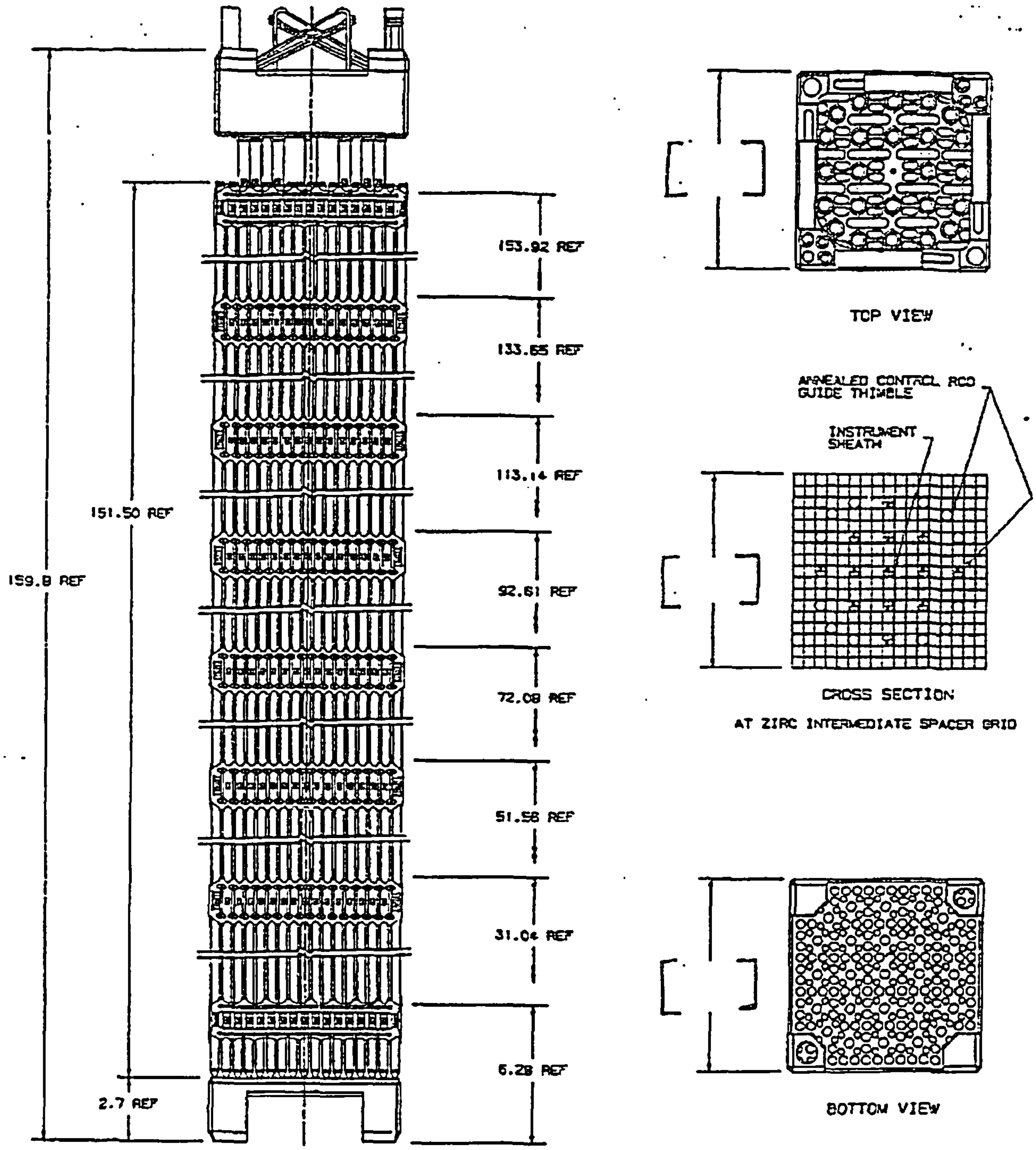


FIGURE 3-1
MARK-BW FUEL ASSEMBLY

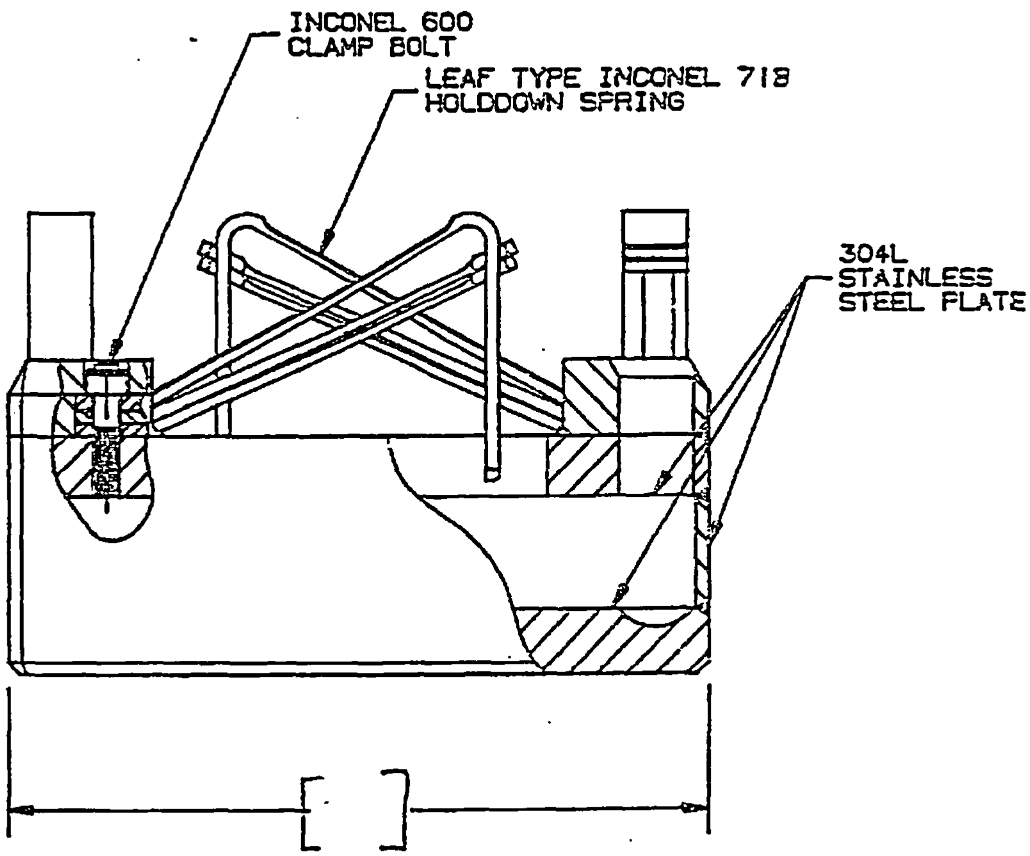


FIGURE 3-2

TYPICAL TOP NOZZLE ASSEMBLY

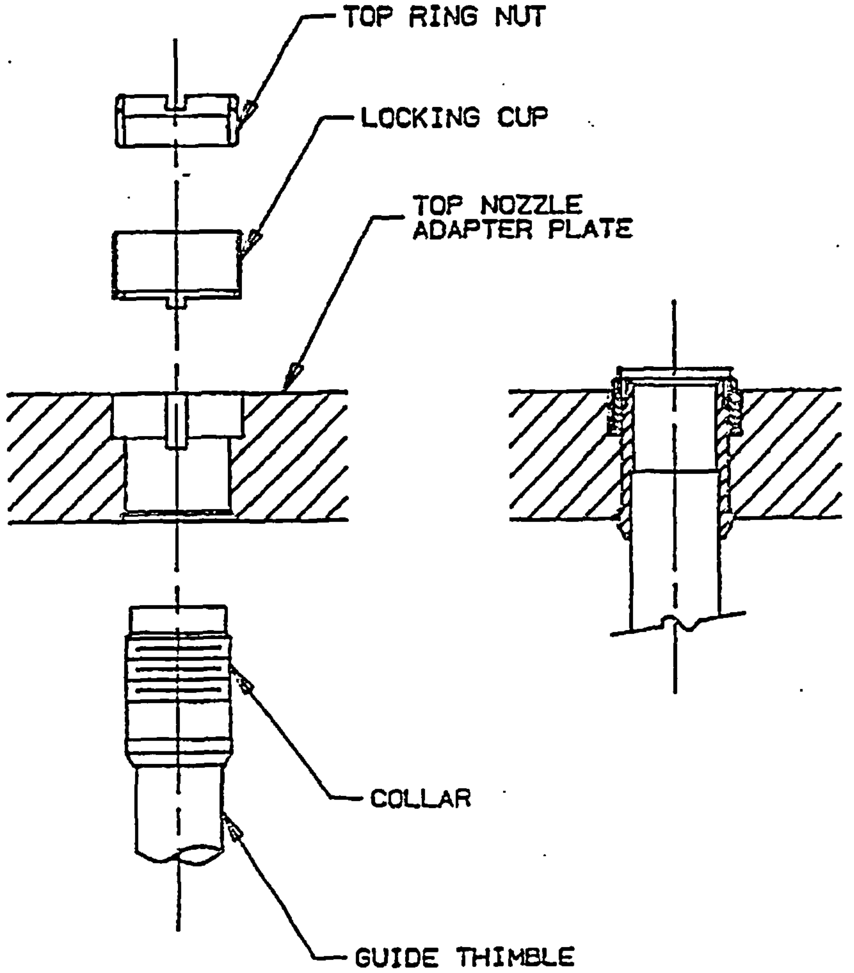


FIGURE 3-3
TYPICAL GUIDE THIMBLE ATTACHMENT TO TOP NOZZLE

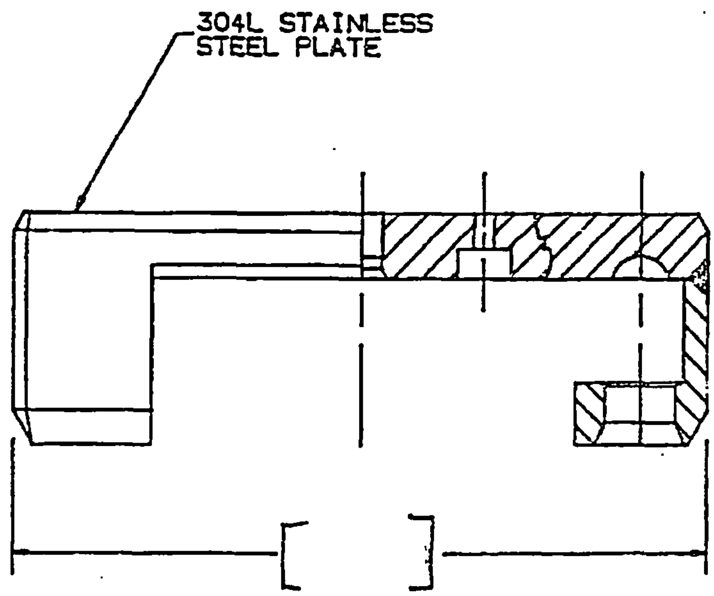


FIGURE 3-4

TYPICAL BOTTOM NOZZLE

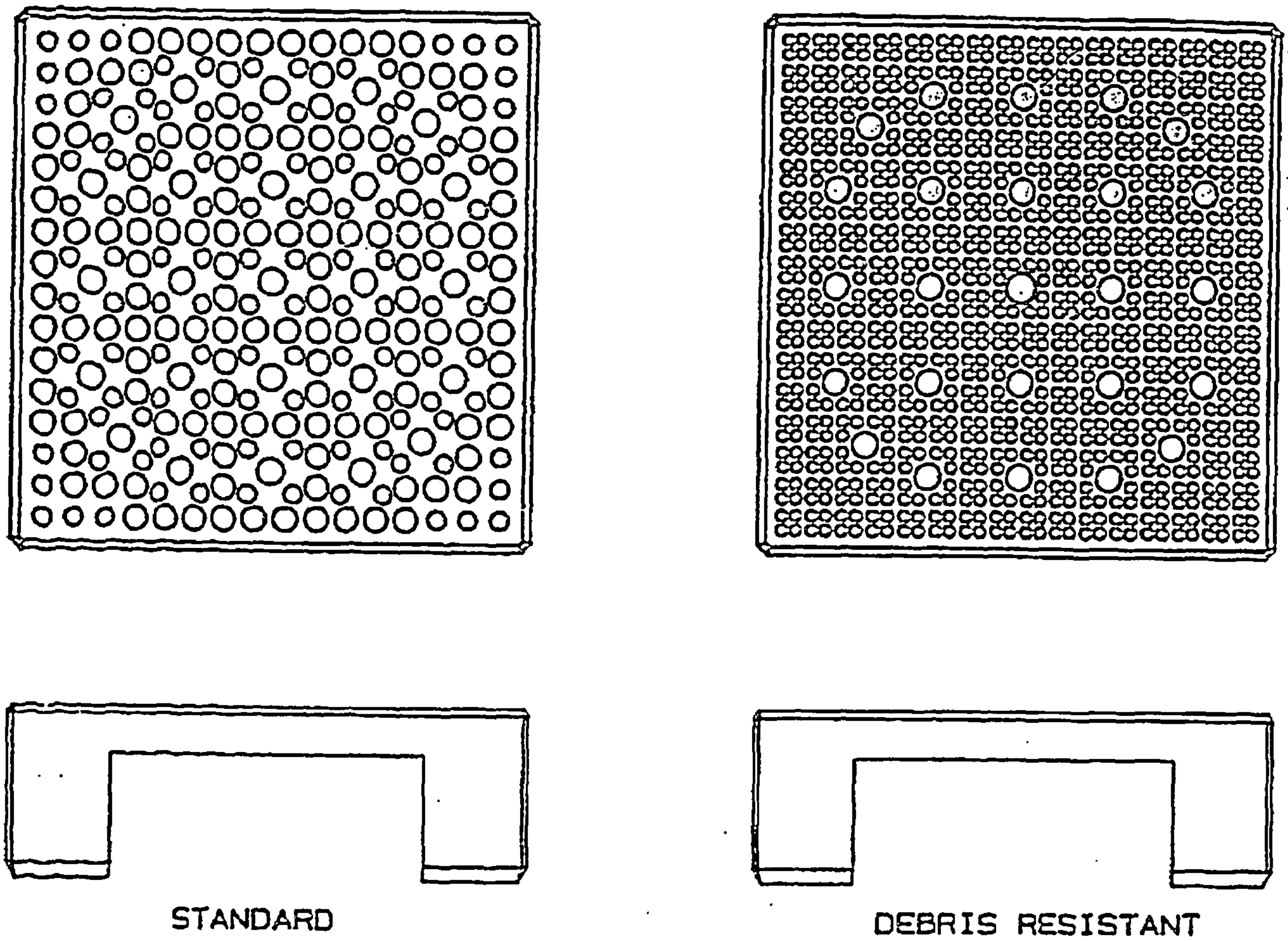


FIGURE 3-5

BOTTOM NOZZLE FLOW HOLE PATTERNS

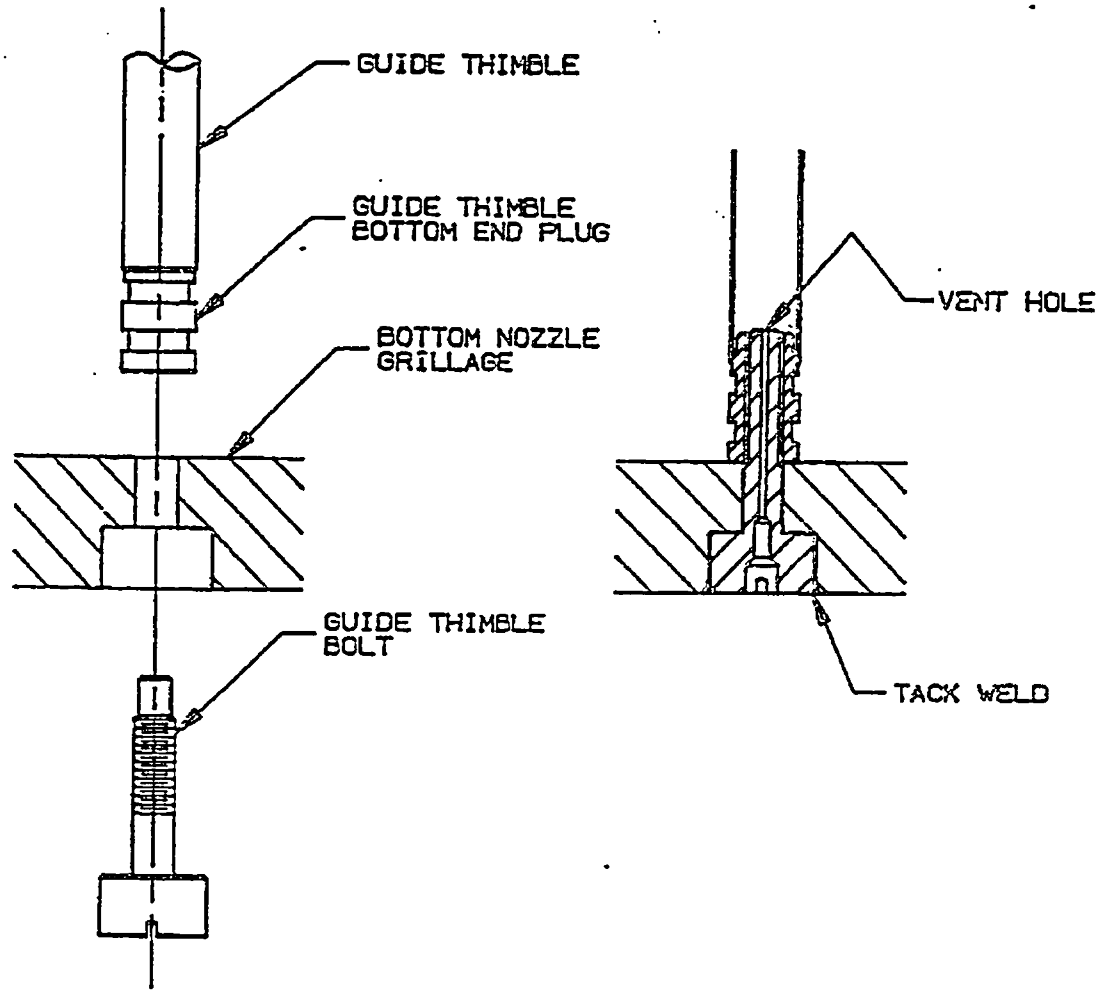


FIGURE 3-6

TYPICAL GUIDE THIMBLE ATTACHMENT TO BOTTOM NOZZLE

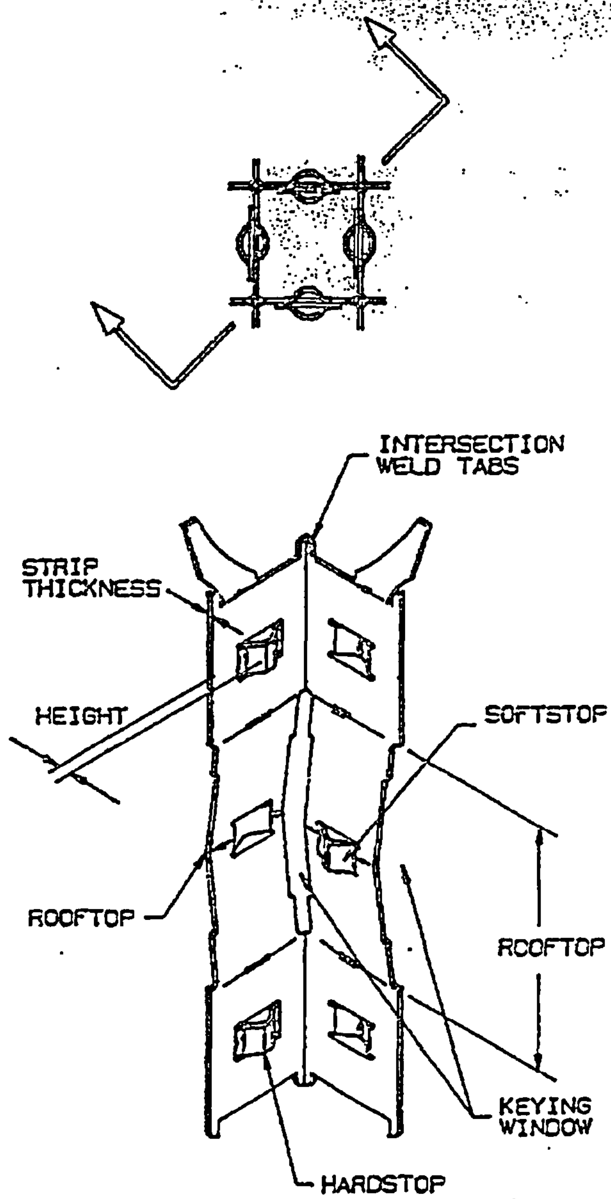


FIGURE 3-7

TYPICAL FEATURES OF A B&W SPACER GRID

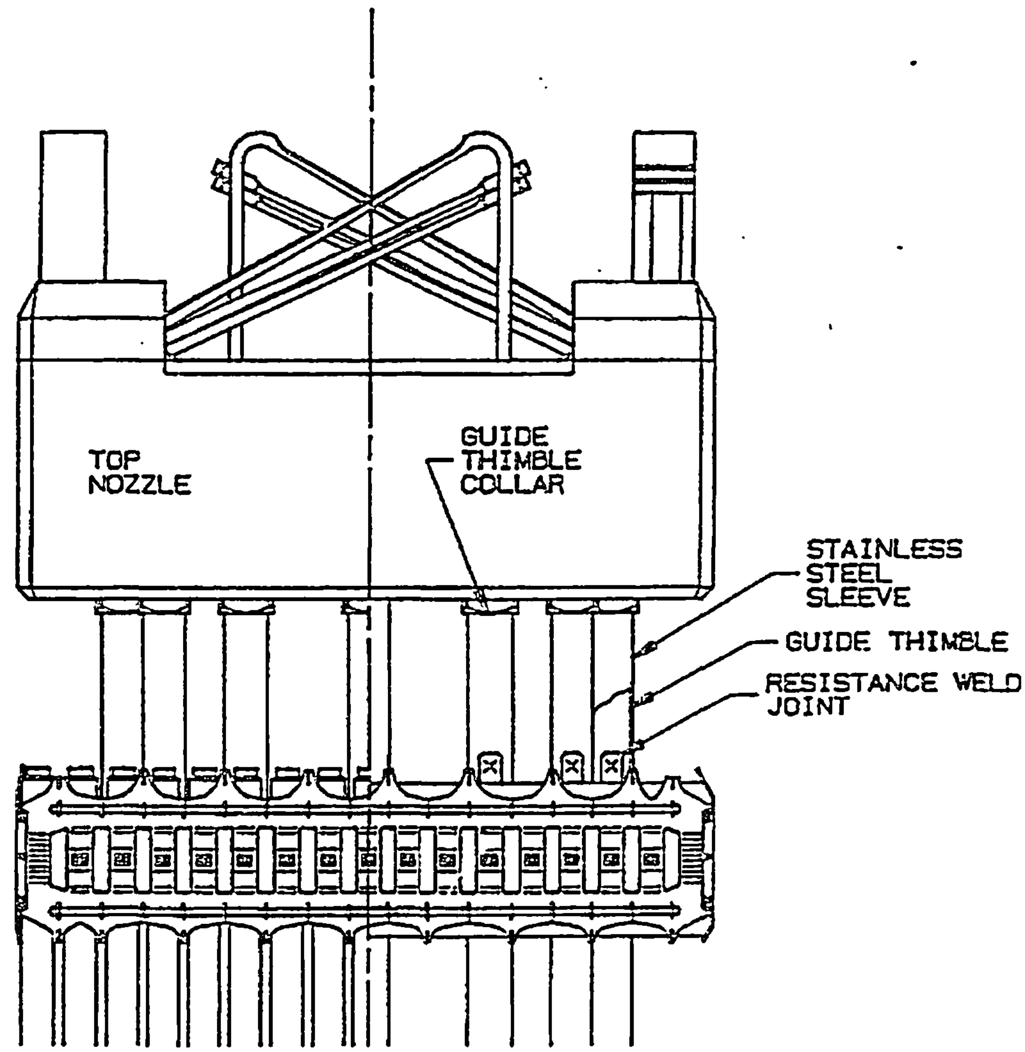


FIGURE 3-8

TYPICAL UPPER END SPACER GRID

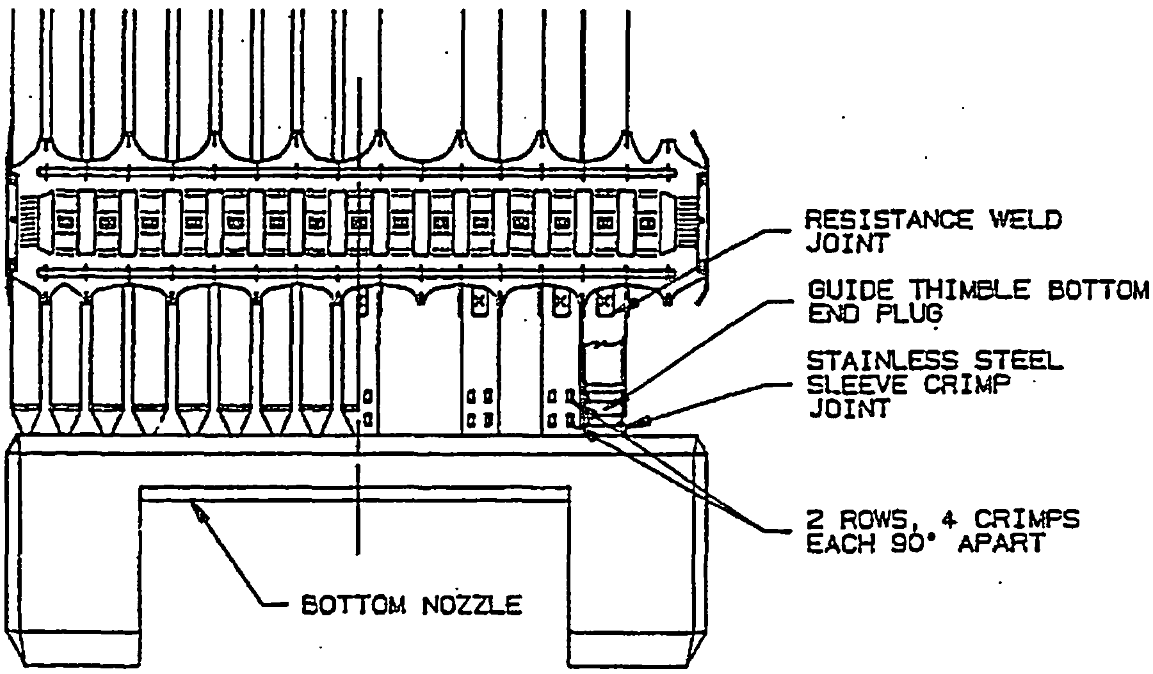
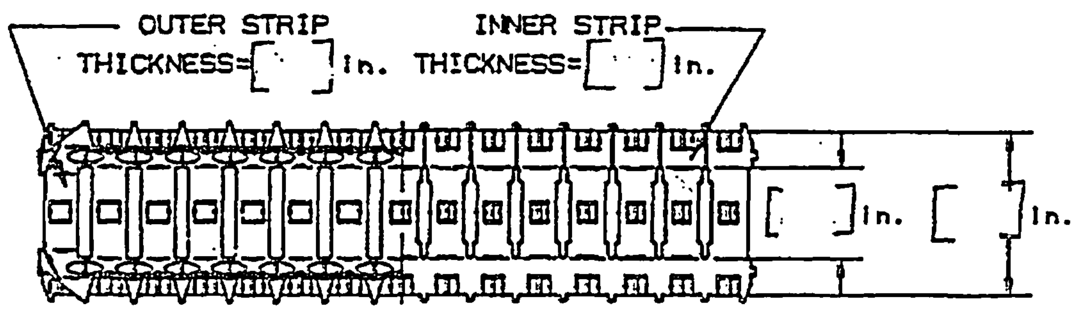
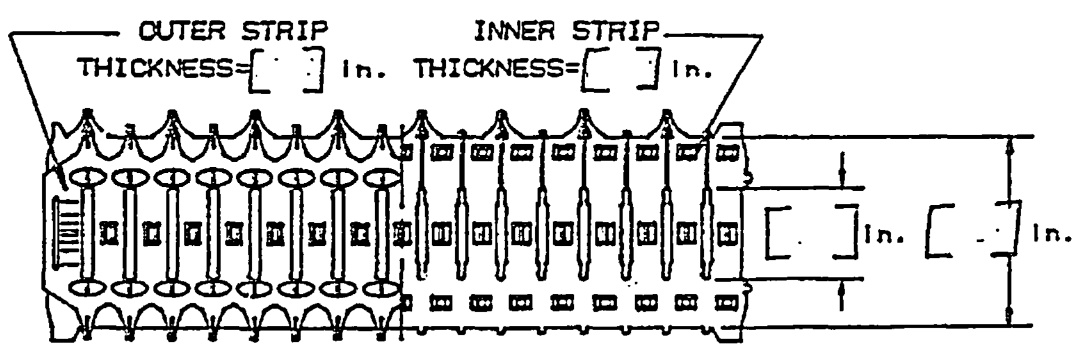


FIGURE 3-9

TYPICAL LOWER END SPACER GRID



MARK-BZ



MARK-BW

FIGURE 3-10

BWFC ZIRCALOY SPACER GRID DESIGNS

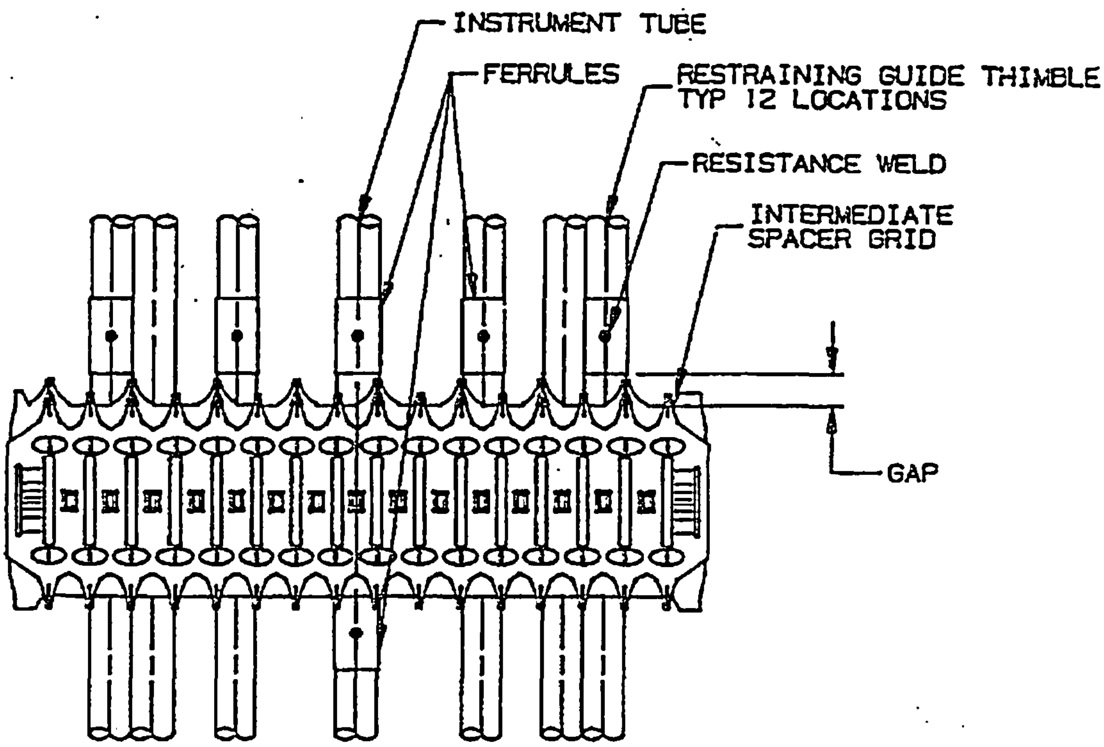


FIGURE 3-11

MARK-BW INTERMEDIATE SPACER GRID RESTRAINT SYSTEM

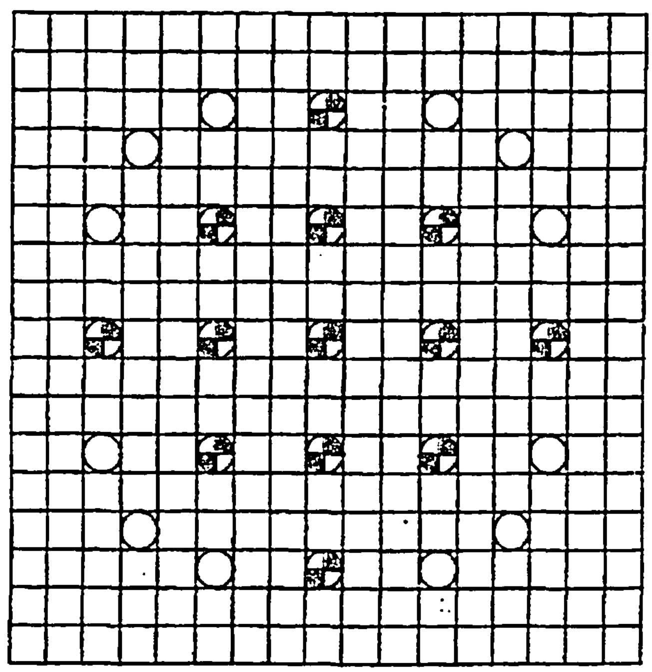


FIGURE 3-12

RESTRAINING GUIDE THIMBLE LOCATIONS

DENOTED BY ⊕

FIGURE 3-13

Typical Fuel Rod Schematic

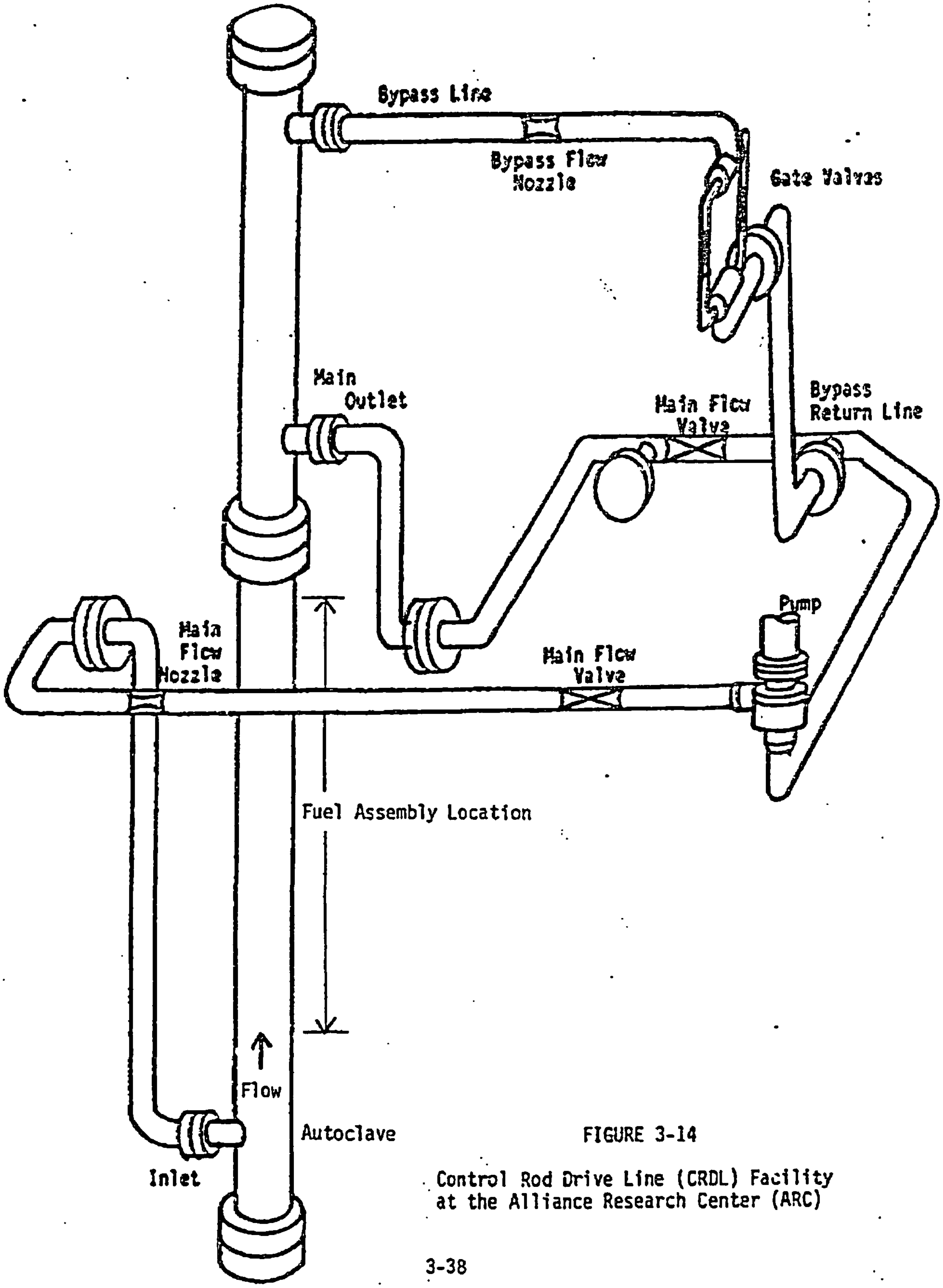


FIGURE 3-14

Control Rod Drive Line (CRDL) Facility
at the Alliance Research Center (ARC)

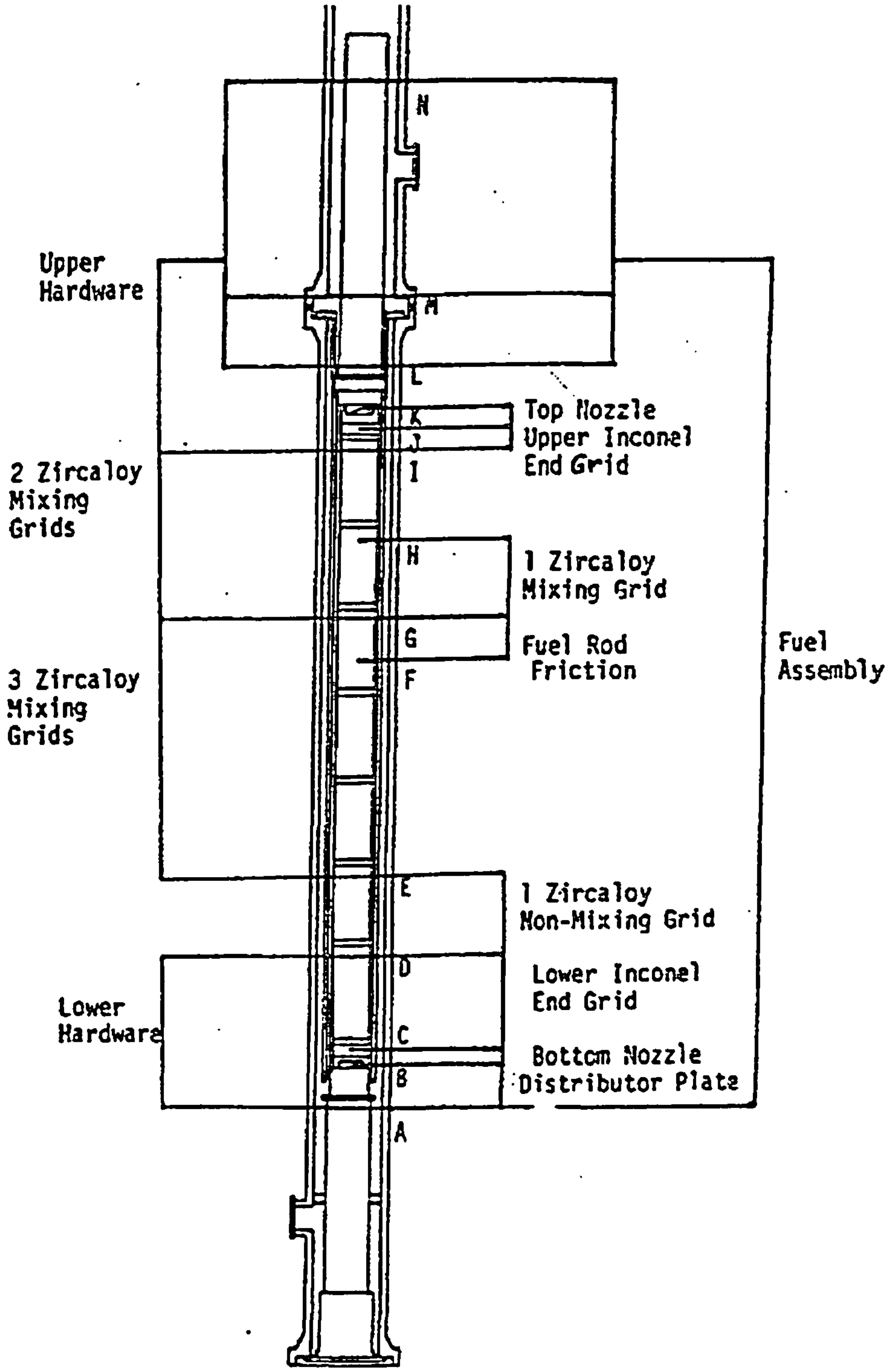
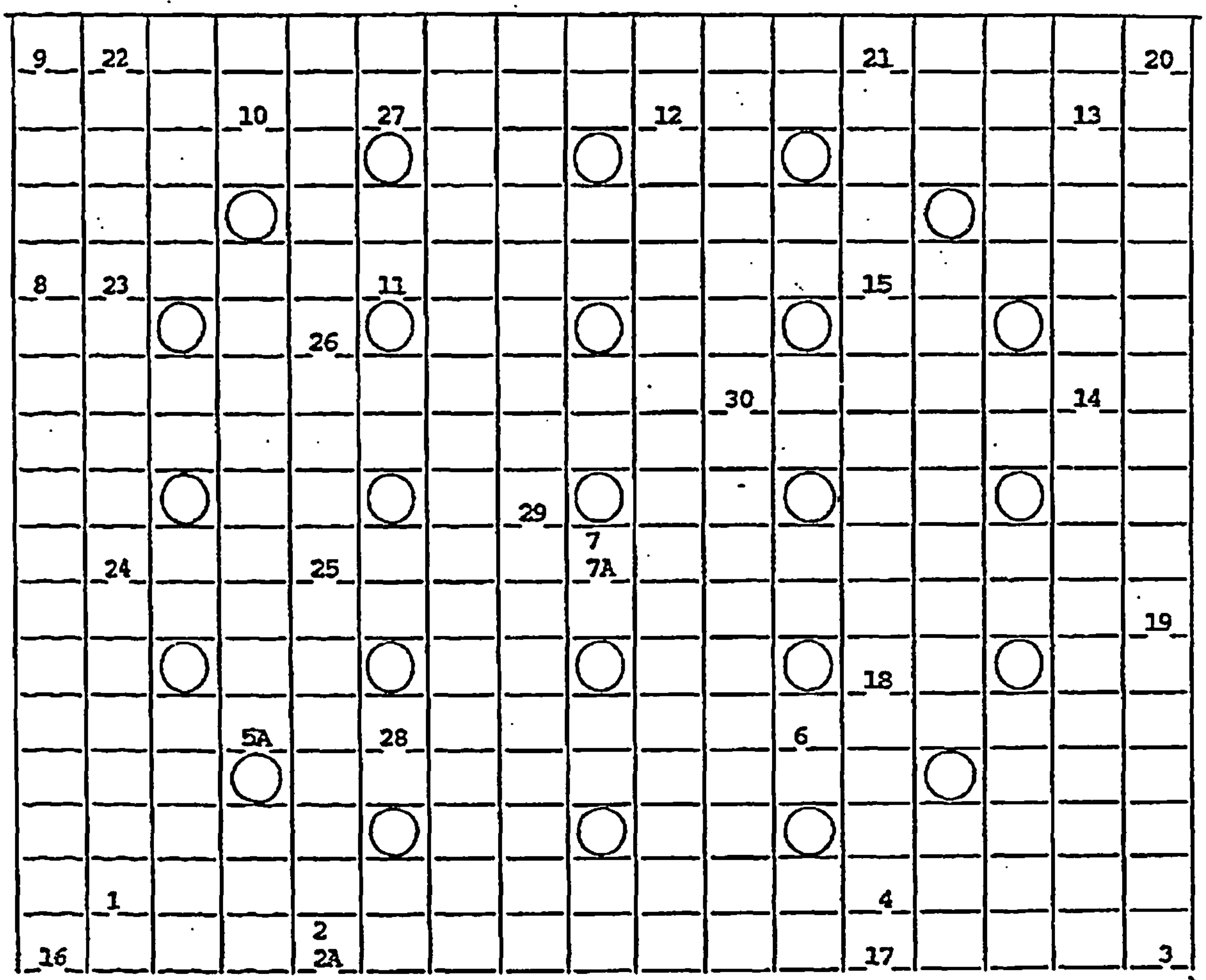


FIGURE 3-15

Pressure Drop Measurement Locations



1 - 15	Removed after 500 hrs. testing.
16 - 30	Removed after 1000 hrs testing.
2A, 5A, 7A	New Rods inserted after first 500 hrs. Test, removed after second 500 hrs. test.

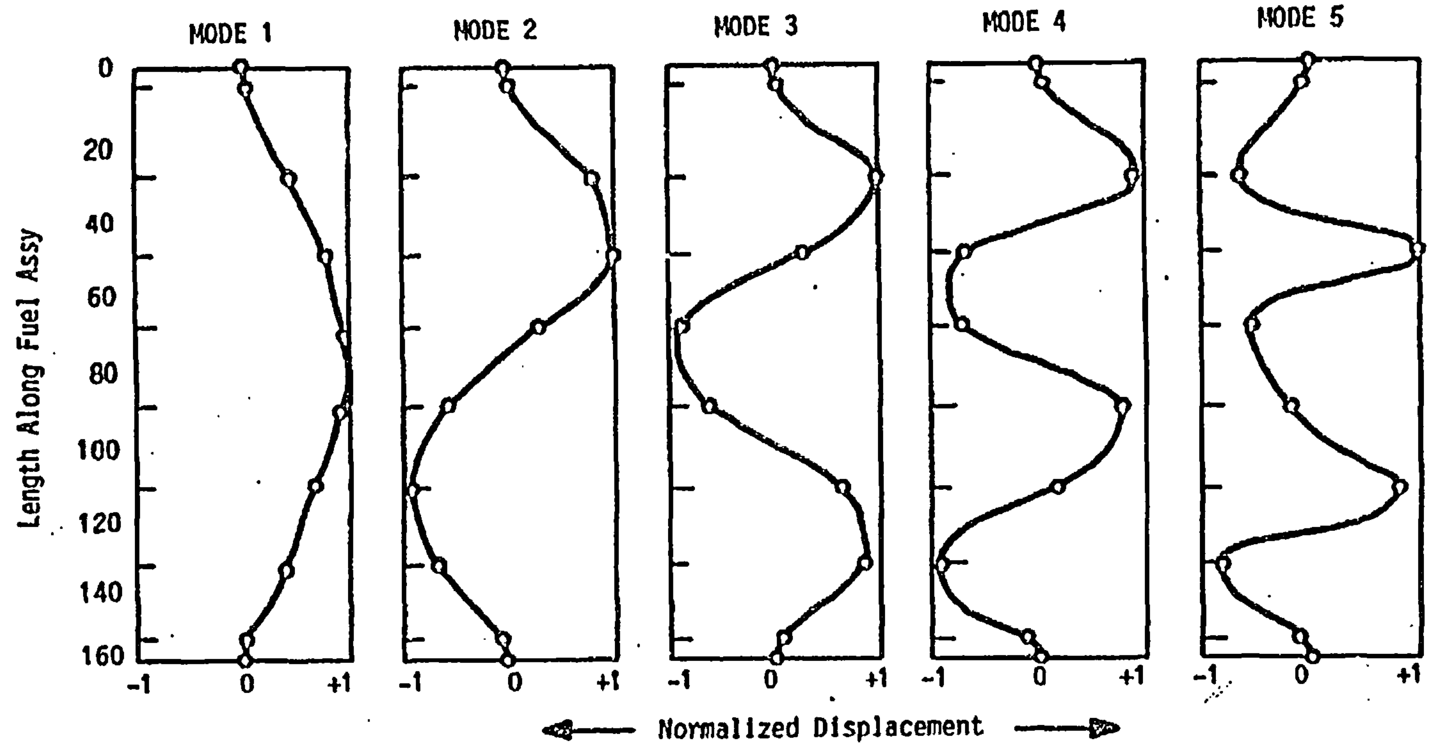
Handling Tool alignment hole (s hole) in Top Nozzle on this corner

FIGURE 3-16

Locations of Fuel Rods Removed for Examination

Figure 3-17. Typical RCCA Trip Curve

Figure 3-18. First Five Mode Shapes of Fuel Assembly



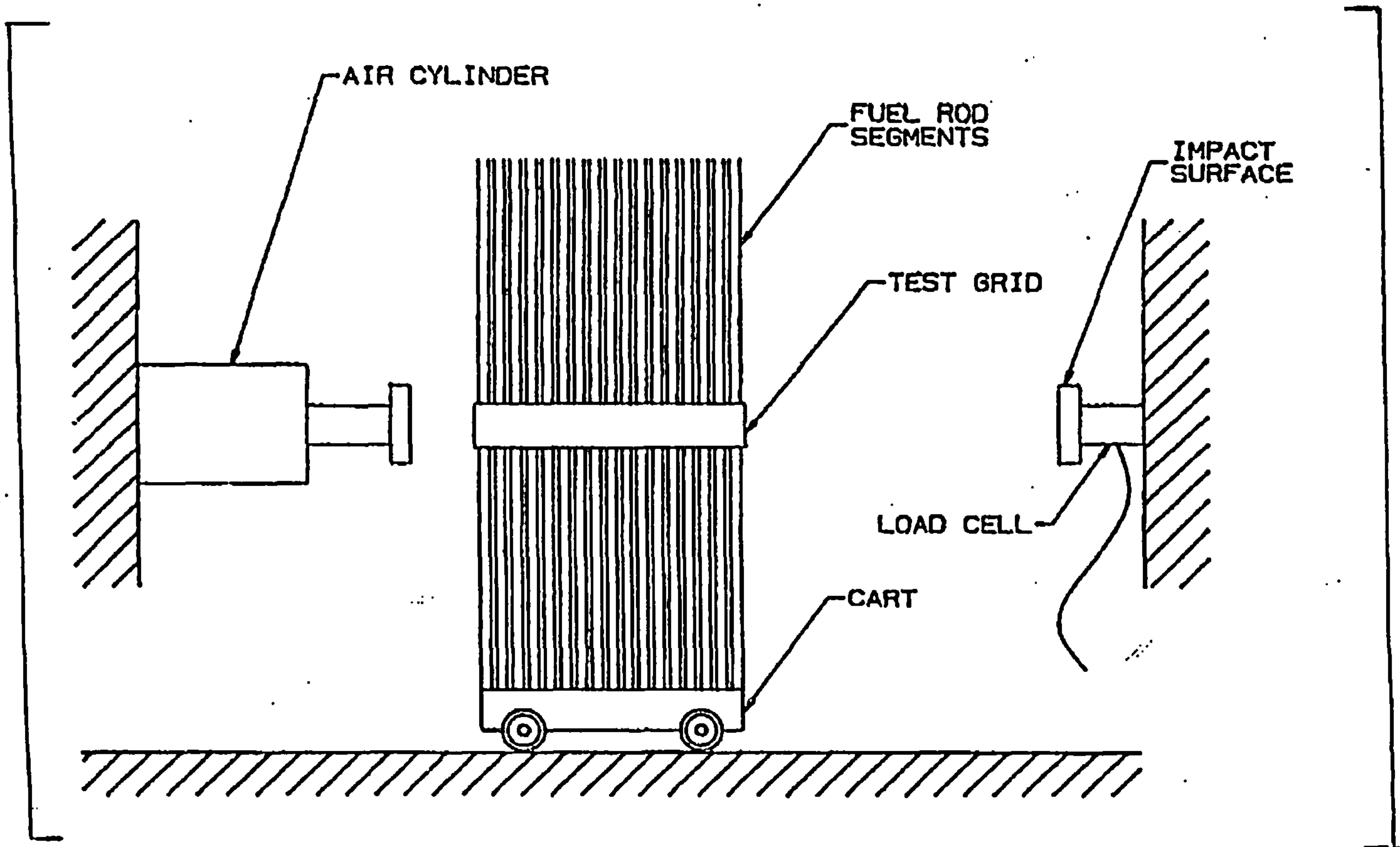


FIGURE 3-19

SPACER GRID IMPACT TEST

Figure 3-20. Guide Thimble Buckling Curves For Span Between Top End Grid and First Intermediate Grid

[ ]

Figure 3-21. Typical Holddown Spring Force/Deflection Curve

4. Mark-BW Fuel Assembly Mechanical Design Evaluation

The Mark-BW fuel assembly design evaluation will be based on the design of the lead assemblies (LAs) with the following differences:

- Spacer grid restraints on twelve guide thimbles instead of four for the LA.

- Thinner grillage on the upper end fitting and a shorter fuel rod to allow more room for fuel rod growth. The UEF grillage has been reduced by approximately 0.2 inch. The fuel rod has been shortened by .15 inches.

Other design changes may be made as options to this design. These changes may include changes in the fuel rod length, the hole pattern in the LEF grillage (debris resistant), and various mechanical fastenings. These product upgrades and other minor evolutionary changes are expected to have only a minor effect on the detailed results presented in this section. However additional analysis will be performed on those changes expected to have any adverse effect on structural integrity or safe shutdown capability. Changes will meet the same requirements and will be evaluated by the same methods presented in this report. Any significant changes will be presented in the reload report for the plant in which the fuel will be used.

In the following section the design basis or criteria will be presented. The method of analyses will be described or referenced, and the results reported. Based on the conservatisms inherent in the analyses, a positive margin is sufficient to assure design acceptability. Margins are determined by the following:

Margin % = [(Allowable - Predicted)/Predicted] x 100 %

4.1 Structural Integrity

To ensure safe and reliable operation the structural integrity of the Mark-BW fuel assembly design was verified for the loadings associated with the normal operation, seismic and Loss Of Coolant Accident (LOCA) events and shipping and handling. The results from the prototype tests and analytical evaluations presented in the following sections verify that the Mark-BW fuel assembly meets the structural requirements for design loads and functions in a safe and reliable manner. The mechanical design bases used for the design are identified in Table 4-1. Section 4.1.1 discusses the normal operation, and evaluation of mechanical compatibility of the Mark-BW fuel design configuration with the Westinghouse resident fuel and Westinghouse designed reactor internals. Section 4.1.2 discusses the analytical methods and the results from the seismic and LOCA evaluation. Section 4.2 discusses the handling characteristics of the Mark-BW design.

4.1.1 Normal Operations

The structural design requirements for the Mark-BW fuel assembly are derived in large part from B&W experience, both in design and in-core operation with similar designs. The design bases and design limits for the Mark-BW fuel assembly are essentially the same as those for the Mark-C, Mark-B, and Mark-BZ designs described in References 1, 2 & 3, except as required to conform to the Catawba/McGuire fuel plant specific design requirements. These requirements are consistent with the " Acceptance Criteria of the Standard Review Plan (NUREG-800) , Section 4.2 (Ref-4), and follow the guide lines established by Section III of the ASME code (Reference 5). Code Level A criteria are used for normal operation and Code Level D criteria are used for LOCA/Seismic. The design bases and evaluation performed to verify the adequacy of the Mark-BW fuel assembly for normal operating conditions are presented in the following subsections.

Table 4-1 shows the stress intensity limits and maximum stress for major structural components for the limiting conditions except seismic and LOCA, which are addressed in Section 4.1.2. The temperature ranges of design transients for the normal operation and the upset condition transients were taken from Table 5.2-1-1 of the McGuire FSAR Volume 4 (Reference 11) except for the temperature ranges, which were modified to include the range 500-650 F for most transients. Table 4-5 provides the transients with the temperature ranges to which the Mark-BW fuel assembly was analyzed for each transient. Low cycle fatigue of the fuel assembly was not a factor in the normal operating analysis since transient cycling involves variation of temperature and pressure which provides negligible stresses.

4.1.1.1 Growth Allowance Design Bases

The fuel assembly to reactor internals gap allowance and upper nozzle to fuel rod growth allowance gap shall be designed to provide a positive clearance during the assembly life time.

Growth Allowance Evaluation

The axial gaps between the top nozzle and reactor core plates and between the top nozzle and fuel rod were conservatively analyzed showing that these gaps allow sufficient margin to accommodate the fuel assembly and fuel rod growth to maximum design burnups established for the Mark-BW fuel assemblies. The target burnups for the Mark-BW design are 55,000 MWd/mtU for the peak assembly, and 60,000 MWd/mtU for the peak rod. The analysis was done by statistically combining the growth tolerances by the square root of the sum of the squares method from nominal and using an upper bound (minimum) gap. The gap between the top nozzle and fuel rod has been increased by approximately 0.3 inches over the gap on the Mark-BW lead assembly by thinning the production top grillage design, and by shortening the fuel rod slightly to allow for additional fuel rod growth for high burnups. Analyses with a finite element model for the upper nozzle has been performed to verify the structural adequacy of the grillage strength for normal, upset and faulted (seismic and LOCA) conditions.

The growth rate for the Zircaloy guide thimbles and fuel rods will be different from that of Westinghouse fuel assemblies. This is due to the different material processing these parts receive as required in B&W specifications.

Irradiation growth of the Zircaloy fuel assembly and fuel rods was conservatively estimated using both EPRI and B&W Post Irradiation Examination (PIE) data. The cooperative DOE/B&W/utility programs at both ANO-1 and Oconee-1 will supply additional Mark-B fuel rod and fuel assembly growth data for high burnups. In addition, an irradiation program to be performed on the Mark-BW lead assemblies, which are in the McGuire Unit 1 reactor during fuel cycles 5, 6 & 7 will provide early burnup performance growth data for the Mark-BW design.

4.1.1.2 Fuel Assembly Holddown Springs

Design Bases

The holddown spring system shall be capable of maintaining fuel assembly contact with the lower support plate during Condition I & II events except for the pump overspeed transient. The fuel assembly top and bottom nozzles shall maintain engagement with reactor internals for all Condition I- IV events. The fuel assembly shall not compress the holddown spring to solid height for any Condition I or II event.

Design Evaluation

The holddown springs were analyzed to show that the holddown springs can accommodate the differential thermal expansion between the fuel assembly and the core internals and irradiation growth. The Mark-BW fuel assembly lift evaluation was performed by comparing the holddown force provided by the fuel assembly leaf springs with the hydraulic forces at various conditions, including the pump overspeed condition. The analysis indicated that the Mark-BW fuel assembly will not lift off under any normal operating condition. At 120 % pump overspeed condition, the fuel assembly will experience some lift-off. The lift-off will be very small and the holddown spring deflection will be less than the worst-case normal operating cold-shutdown condition. The fuel assembly lift-off will cause no additional plastic set of the holddown spring. The Mark-BW fuel assembly holddown spring stress calculations showed that the holddown spring is structurally adequate under all static and fatigue loading conditions. The design of the Mark-BW holddown spring is the same as that of the NFI 17x17 fuel design, which has performed well in five reactors in Japan (Table 4-9). The top nozzle provides positive retention of the holddown spring in case of the unlikely event of spring breakage. The analysis also determined that the upper core plate guide pin can not bottom out in the guide pin hole of the upper nozzle during the fuel design life.

The clamp screws which mount the holddown springs on the top plate of the top nozzle were analyzed for normal operation and the fatigue loads. The analysis showed that the clamp screws are adequate for all loading conditions.

<u>4.1.1.3 Guide Thimble</u>

Design Bases

Thimble buckling shall not occur during normal operation (Condition I) or any transient condition where control rod insertion is required by the safety analysis.

Design Evaluation

Various loading situations relative to guide thimble buckling and ferrule retention were analyzed. Under hypothetical " worst case " type of loading condition (120 % pump overspeed) a [ ] margin against buckling exists. As previously described, a conservative buckling test was run on a full scale single span of a guide thimble (two grids and a guide thimble segment) at room temperature. This buckling load was corrected for corrosion, tolerances and temperature effects. The margins of safety reported in Table 4-4 for the various guide thimble normal operating buckling loading conditions are based on the following two bases.

1) Load to produce a midspan lateral deflection of [ ] inch using the secant formula as discussed in Section 3.3.5.5, which is based on not affecting control rod insertion or trip performance.

2) Load to produce maximum allowable compressive yield stress (Sy) using the secant formula as discussed in Section 3.3.5.5.

The margins are reported for cases 1) 100 % full power (FP) mechanical design flow rate, 2) 120 % FP mechanical design flow rate (pump overspeed condition), and 3) 100 % FP mechanical design flow rate with an upper bound scram load of [ ] lbs (which is based on assuming control rod spring fully compressed during a scram). For the first two cases, the margins reported are based on the first basis specified above, as the control rod insertion requirement has to be shown for these cases. For the third case, the margin reported is based on the second basis specified above, since after a SCRAM, the control rod is fully inserted in the guide thimble and hence the control rod insertion is not a concern for this case. All margins were shown to be acceptable.

The ferrule to Zircaloy spacer grid interface was tested to determine the stiffness and strength (maximum load) of this interface. The results of this test, together with the results from the guide thimble buckling test were used in an evaluation of the intermediate spacer grid restraint system. The evaluation showed that sufficient margins exist for ferrule/grid interface strength under all expected conditions. All margins were shown to be acceptable. A complete qualification of the ferrule welding process for production on the Mark-BW fuel assemblies is in progress. Testing results of the resistance welds of the ferrules to Zircaloy guide thimble segments obtained to date, indicate that the weld will provide adequate strength during exposure to the reactor environment.

4.1.1.4 Spacer Grid

Design Bases

The design bases of the top and bottom Inconel grids and the intermediate Zircaloy grids remain the same as the Mark-BZ design, Ref-3, which require that no crushing deformations occur due to normal operation (Condition I) and Condition II event loading, and that these grids provide adequate support to maintain the fuel rods in a coolable configuration for all conditions.

Design Evaluation

The top and bottom Inconel grids of the Mark-BW fuel assembly are the same as those of the Standard NFI fuel design. As discussed in Section 3.1 the intermediate Zircaloy spacer grid design is similar to the B&W 15x15 Mark-BZ spacer grid design. The structural integrity of the intermediate Zircaloy spacer grid was confirmed through structural tests summarized in Section 3.3.5. Calculations were performed to show the acceptability of the fuel rod restraining forces provided by the grid. The structural performance of the grid under seismic and LOCA loadings as described in Section 4.1.2 has shown that the Zircaloy spacer grid provides adequate design margins.

4.1.1.5 Interface with Adjacent Assembly

Design Bases

As a minimum, the flat vertical surfaces of grids on adjacent assemblies (including residual Westinghouse fabricated fuel assemblies) shall have a positive overlap at the " worst case " conditions. A minimum of [ ] MWd/mtU differential burnup between adjacent assemblies shall be considered.

Design Evaluation

Elevations of the grid have been selected by adjusting the spacer grids and ferrules to maximize axial alignment with the grids of the adjacent fuel assemblies. Grid to grid contact is maintained at all times between adjacent fuel assemblies. This is necessary to prevent fuel rod wear by grids on an adjacent fuel assembly and to maintain the fuel assemblies structural capability during seismic and LOCA conditions.

4.1.1.6 Fuel Rod Fretting and Wear

Design Bases

The fuel assembly design shall be shown to provide sufficient support to limit fuel rod vibration and clad fretting wear.

Design Evaluation

A life and wear test was conducted at maximum reactor flow conditions for more than [ ] hours to evaluate the fretting characteristics of fuel rods and spacer grids. Results of the test showed no indication of adverse fretting wear or progressive wear of fuel rods. Details of the fretting corrosion testing are described in Section 3.3.3.

Operational experience is also available from the Mk-BZ Lead Test Assembly (LTA). The Mk-BZ LTA is a Mark-B fuel assembly with Zircaloy Intermediate spacer grids which was irradiated to a burnup of [ ] GWd/mtU. A poolside examination was made of fuel rods removed from the MK-BZ LTA (reference 27). The Zircaloy intermediate spacer grids for the Mark-BW were based on this design. A total of five rods were pulled from the two of the four MK-BZ LTAs. All spacer grid stop-fuel rod contact sites were examined. No evidence of fretting was visible.

4.1.1.7 Rod Bow

Design Bases

Fuel rod bowing shall be evaluated with respect to the mechanical and thermal/hydraulic performance of the fuel assembly. The fuel assembly shall not exhibit excessive fuel assembly bow during its operation life.

Design Evaluation

There are several features of the Mark-BW fuel assembly design that improve its fuel rod bow performance. These features include not rigidly attaching the intermediate spacer grids to the guide thimbles and using a keyable spacer grid design. The net result is that the axial strains induced in the cladding are reduced which translates to lower rod bow. Similarities between the Mark-BW and other B&W fuel designs (Mark-B, -BZ,-C) permit the use of the rod bow prediction as presented in the rod bow topical report (BAW-10147, Ref 6) in the thermal-hydraulic evaluation discussed in Section 5.4 for plant operation limits. This report has been approved by the NRC for licensing previous B&W fuel designs.

The fuel rod bow performance of Zircaloy grids has been verified by poolside PIE (reference 28). Water channel measurements made on the Mk-BZ LTA following irradiation to [ ] GWd/mtU showed the fuel rod bow was enveloped by the predictions from the fuel rod bow topical.

4.1.1.8 Top Nozzle and Bottom Nozzle

Design Bases

The top nozzle and bottom nozzle design bases are the same as given in Ref 1, Section 4.2 which are based on the ASME Boiler and Pressure Vessel code, Section III limits.

Design Evaluation

Finite element analyses of the top nozzle grillage and the bottom nozzle grillage have been performed to show that the designs are more than adequate to withstand the normal operating loads. The analyses were performed using enveloping conservative operating loads. Particularly, the loads during shutdown were evaluated at 100 °F, EOL at which the holddown force is maximum. Also at the operating condition of 650 °F a conservative SCRAM load was applied in addition to the holddown force. For the lower nozzle, in addition to the SCRAM load, the weight of the assembly was also applied to the lower nozzle grillage.

For the bottom nozzle, the finite element analyses discussed above were performed for the lead assembly. Currently, the debris resistant bottom nozzle design as described in Section 3.1 is being pursued for the production design. Preliminary calculations have shown the adequacy of the debris resistant bottom design for the normal operating loads. This will be confirmed by finite element analyses scheduled for completion by the later part of 1988.

<u>4.1.1.9 Control Rod trip Times</u>

Design Bases

The fuel assembly shall not experience any permanent deformation during normal operation (Condition I) or a Condition II event that would cause the control component drop time to increase beyond the following limit.

Maximum Control Rod Trip Time: [ ] seconds: as measured from the start of control rod spider movement to dashpot entry by the control rod.
[ ] seconds: to be used in the accident analysis.

Design Evaluation

The diameters of the Mark-BW guide thimbles are identical to those of the Standard Westinghouse design at the upper and dashpot sections.

4.1.1.10 Mechanical Compatibility

Design Bases

The Mark-BW fuel assembly shall be dimensionally and hydraulically compatible with the Westinghouse resident fuel and dimensionally compatible with other core components and fuel handling equipment.

Design evaluation

The Mark-BW fuel assembly design data are given in Table 3-1. In this table, comparisons are made between both the Westinghouse STD and the OFA fuel to show design similarities and differences. Figure 3-1 shows a Mark-BW full length schematic view. The Mark-BW fuel assembly has the same cross-sectional envelope as the 17x17 Westinghouse fuel assembly. Mechanical interaction between fuel assemblies is confined to the grid locations. The grid elevations of the Mark-BW fuel assembly match the 17x17 Westinghouse fuel assembly, minimizing the effects of mechanical and hydraulic interaction between assemblies. The Mark-BW fuel assembly is designed to be fully compatible with the STD and OFA Westinghouse 17x17 fuel assembly design, reactor internal interfaces, fuel handling and refueling equipment, and spent fuel storage racks. Compatibility of the existing control components with the Mark-BW design is assured by the similarity of critical dimensions such as guide thimble locations with those used in the Westinghouse STD and OFA designs.

4.1.2 LOCA and Seismic Loading

The following criteria have been established for the fuel assembly seismic and LOCA analysis which are consistent with the "Acceptance Criteria" of the Standard Review Plan (NUREG-0800), Section 4.2.

1) For Operational Basis Earthquake (OBE)

The fuel assembly is designed to ensure safe operation following an OBE.

2) For Safe Shutdown Earthquake (SSE)

The fuel assembly is designed to allow control rod insertion and to maintain a coolable geometry.

3) For Loss of Coolant Accident (LOCA) or Combined LOCA plus SSE.

The fuel assembly is designed to allow for the safe shutdown of the reactor systems.

In the accident analyses, the lateral effect (LOCA and seismic) and the vertical effect (LOCA) are investigated separately. This leads to a development of a lateral model representing a row of assemblies located on a symmetry axis of the core and a vertical model of the fuel assembly. This section gives a description of these models and discusses the results of the analysis. Only the LOCA effect was analyzed in the vertical direction, as the seismic excitation in this direction will cause fuel assembly liftoff.

4.1.2.1 Fuel Assembly Lateral Analysis

The seismic and LOCA time history motions of the upper and lower core plates and the barrel at the core plate elevation were provided by Duke Power Company for each of the McGuire and Catawba nuclear stations. These motions were applied to the reactor core model as shown in Figure 4-1. The fuel assembly deflection and grid impact force responses were determined using the general procedure outlined in B&W topical report BAW-10133P, Rev 1 (Reference 7), except for a minor conservative modification that was made to the reactor core model. This report has received the NRC approval for referencing in licensing applications. In the reactor core model, the fuel assembly was represented by [ ] masses instead of [ ] masses as described in the Reference 7 report. It has been shown that [ ] masses in the core model result in the highest load.

A brief description of the analytical core model and the results of the analysis are summarized below. Details of the core model and the analysis method may be found in Reference 7.

The core model contains [ ] fuel assemblies in a single row to simulate the most severe impact loading condition for the fuel assemblies. A parametric study indicated that the [ ] fuel assembly model gives the highest impact loading condition and as the number of assemblies increases beyond five, the load decreases. Each fuel assembly was modelled as a spring mass system, with six masses lumped at the intermediate spacer grid locations and seven beam elements representing the fuel rod and guide thimble cross-sectional inertias. The values for the rotational springs and water mass distribution were calculated based on the fuel assembly experimentally determined mode shapes and corresponding frequencies. The assemblies were interconnected by [ ] gaped springs, that represent the spacer grid impact properties, such as impact stiffness, damping, etc. These properties were obtained from the spacer grid impact test results. Calculated nominal operating temperature gaps between the adjacent fuel assemblies and between baffle plate and peripheral fuel assemblies were used in the analysis. The SSE and LOCA calculation in the analysis was based on fuel assembly properties at 600 F and full reactor coolant flow condition.

The upper core plate, lower core plate and the core barrel motions at upper core plate elevations were simultaneously applied to the complete core model. The core barrel motions at each individual grid elevation were then linearly interpolated between the motions of the upper core barrel and the lower core barrel at every particular time instant. The model was analyzed using the computer program STARS version 15.1, described in Reference-8. The STARS program performs direct numerical integration of the equation of motion subjected to known displacement time histories.

The static and dynamic characteristics of the Mark-BW fuel assemblies were determined analytically and experimentally. These characteristics were found to be compatible within the range of values provided by Westinghouse for their resident fuel assemblies. Also both Westinghouse and Mark-BW spacer grids have similar overall stiffness and geometric buckling characteristics. The typical grid buckling failure mode obtained from the Conn-Yankee spacer grid (Westinghouse 15x15 Inconel grid - test done by B&W Fuel Co.) and the Mark-BW spacer grid impact tests was a racking failure mode for both type of the grids. Racking was confined to the first [ ] rows on the impact surface side of the grid, and the guide thimble positions were not altered. In view of these similar results on the Westinghouse and the Mark-BW fuel assembly and the spacer grid characteristics, the Mark-BW analysis results are applicable to the Westinghouse fuel.

For the Safe Shutdown Earthquake, the calculated spacer grid impact loads are within the elastic load limit. Therefore for the Safe Shutdown Earthquake (SSE), the requirement of a control rod insertion and coolable geometry is met. As the spacer grid impact loads are within the elastic load limit for the SSE, these results also satisfy the Operational Basis Earthquake (OBE) requirements, that the assembly or component not exceed its yield limit (usually the magnitude of the OBE is half the magnitude of the SSE). Hence a separate OBE analysis is not required.

For the LOCA loading, a very few of the spacer grid impact loads slightly exceeded the spacer grid elastic load limit. This load on the spacer grid resulted in minor plastic deformation. However, calculations showed that the fuel rods remained in a coolable geometry and the guide thimble positions were not altered. Thus, the insertion of the control rod will not be hindered. These calculations were based on grid cell geometry and the NRC's definition of a fully collapsed grid, (Reference 4) all soft stops fully compressed and fuel rod in hard contact with all hard stops. For this mode of deformation the reduction in subchannel flow area was [ ]%, which correspond to a [ ] inch deflection of the grid. This is consistent with results of Mark-BZ and Inconel grids, which have shown compliance with NUREG-0800, Section 4.2. The ECCS analysis will verify that for the collapsed grid, the fuel rod cladding temperature is within the acceptance limit of 2200 °F. The ECCS LOCA analysis will be reported in BWFC Topical Report-BAW-10174P which is scheduled to be issued by the end of calendar year 1988. The loads for LOCA plus SSE were combined by the sum of the square-root-of the squares method as discussed and accepted by the NRC in Ref 4, NUREG-0800.

The results of the maximum spacer grid impact forces for the faulted conditions are summarized in Table 4-2. The maximum grid impact force of 4888 lbs occurred at the middle intermediate grid location of the peripheral fuel assembly. Figure 4-5 shows the typical impact force plot for the peripheral fuel assembly, where the maximum grid impact load occurred. Only one impact above the elastic impact load was experienced for the peripheral fuel assembly position. The subsequent impact was of much lower magnitude and will not have a significant effect on the magnitude of the grid deformation.

The grid permanent deformation was measured as a function of impact load during the dynamic impact tests as discussed in Section 3.3.5.1. An upper bound limit curve of the grid permanent deformation was obtained from the test data at room temperature and at reactor operating conditions. The reactor operating temperature curve was used to obtain the spacer grid permanent deformation from the impact force obtained from the analysis. The coolable geometry of the fuel assembly corresponding to the maximum calculated grid deformation was confirmed by performing the limiting grid deformation calculation as discussed above.

4.1.2.2 Vertical LOCA Analysis

For the vertical LOCA response analysis, a modification to the fuel assembly axial model, as described in BAW-10133P, Rev 1 (Reference 7) was required, because of the design differences between the Mark-C and the Mark-BW designs in the method of restraining the spacer grids. This is a minor change in only the mechanics of the analysis.

The finite element vertical model of the fuel assembly as shown in Figure 4-2 is a lumped mass model with one dimensional spring and sliding friction elements just like the Mark-C fuel assembly vertical model described in Reference 7. The model was analyzed using the general finite element code ANSYS (Reference 9) because of its capability to distribute a given hydraulic force time history over a large number of mass nodes. The spring element characteristics were directly determined from geometric and material considerations, except for the nozzles whose stiffness was obtained from the finite element analysis. The sliding element friction characteristics were obtained from slip load tests on production grids. Non-linear effects such as holddown spring pre-load, fuel to grillage gap, and fuel rod to spacer grid friction were incorporated in this model. The spring rate of the restrained and the unrestrained guide thimbles and the fuel rod were determined from their respective cross sectional areas of the cladding, and the span length. A concentrated mass was added to the spring elements at appropriate nodal locations for the dynamic analysis.

The hydraulic loads on the reactor core during a LOCA were determined by B&W from their CRAFT analysis (for methods refer to Reference 10) for the McGuire and Catawba Nuclear Stations. The following design cases were considered.

Case-1. A break at the surge line nozzle attachment to the hot leg.
Case-2 A break at the safety injection line nozzle attachment to the cold leg.

Figures 4-3 and 4-4 show the reactor core forcing functions respectively for the above cases. The fuel assembly force time history applied to the fuel assembly vertical model was obtained by normalizing the 193 assembly core to a single assembly basis. The hydraulic force on the assembly was applied to the ANSYS model as a force time history input. LOCA distribution factors for the fuel assembly components were calculated based on the form loss coefficient associated with each node. Analyses were performed for both beginning and end of life values to determine the worst case loading condition.

The component forces obtained from the analysis are summarized in Table 4-3. The results of the analysis show that because of the holddown spring applied preload and stiffness, the fuel assembly during the LOCA does not contact the upper core plate. These forces are well below conservatively calculated allowable loads for the guide thimbles and the fuel rods.

4.1.2.3. Fuel Assembly Structural Analysis

The fuel assembly component stress analysis was performed using the loads generated by the seismic and LOCA analysis. There were two load cases that could be analyzed- SSE and SSE plus LOCA. The criteria for LOCA and SSE plus LOCA are the same. Because of this, only the SSE plus LOCA was analyzed since it produced the highest loads.

The axial and lateral loads obtained from the seismic and LOCA loads were used for the subsequent stress analysis. ASME Code- Appendix F (Ref-4) was used as a guide for the fuel assembly general stress criteria. The analysis for most components used classical techniques. In some cases, failure loads as established by testing were incorporated per the ASME Code. In most cases, the adequacy of the components was shown using the nominal dimensions. The upper and lower nozzles were analyzed using the finite element code ANSYS (Reference 9). The guide thimble buckling analysis was performed using the column secant formula as given by the Equation in Section 3.3.5.5. The guide thimble buckling limit used in the analysis is conservative since the guide thimble will deflect and contact the fuel rod which provide support and a higher allowable load. A load factor of [ ] was used to account for unequal loading due to external factors, fabrication differences and inherent design factors. The guide thimble and fuel rod stresses resulting from the maximum probable fuel assembly deflection were evaluated. The fuel assembly maximum probable deflection was calculated using the accumulated fuel assembly gaps of the rows having the maximum number of fuel assemblies across the core diameter. This is the maximum deflection allowed by the reactor internals constraint system (core baffle plates). The total stress intensity in fuel rods was calculated by considering contribution from 1) dynamic bending loads 2) dynamic axial loads and 3) steady state hoop stress caused by the pressure differential between the reactor system pressure and fuel rod internal pressure. The internal pressure considered was the minimum fill gas pressure at room temperature before operation. This is conservative in the determination of maximum stress intensity in the fuel rod, since the internal pressure considered was a minimum value.

The maximum stresses for various fuel assembly components are summarized in Table 4-4. They indicate that all major components of the fuel assembly meet the design requirements for the SSE and a combined SSE plus LOCA event with adequate structural margin. The results of the SSE analysis meet the design criteria for OBE described in Section 3 of topical report BAW-10133P, Rev 1 (Reference-7). So separate OBE stress results are not required. Also the SSE requirement of control rod insertion was fulfilled for a combined SSE plus LOCA and therefor provides added conservatism to the analysis.

4.2. Fuel Assembly Mechanical Compatibility and Handling Characteristics

Mechanical compatibility of the Mark-BW fuel assembly with the reactor internals, handling and storage equipment, and resident fuel assemblies has been confirmed by direct comparisons with the dimensions of Westinghouse fuel assemblies affecting these critical interfaces. Many of these critical interface dimensions were supplied to Babcock & Wilcox by Westinghouse Electric Corporation through Duke Power Company. The information included outline drawings containing envelope dimensions, spacer grid locations, and critical interface dimensions of the top and bottom nozzles.

The Zircaloy intermediate spacer grids of the Mk-BW fuel assembly include several design features of the B&W Fuel Company's Mk-BZ spacer grid which promote resistance to hanging-up with other fuel assemblies or equipment. The leading edges of the exterior strips of the spacer grid assemblies are inboard of the plane of the outside edges of the peripheral fuel rods. Each intermediate spacer grid assembly is checked for hang-up resistance on the fuel assembly by sliding a straight edge along the plane of the outside of the peripheral fuel rods and over the spacer grid. Lead-in is also provided at the corners of the grid to resist hang-up. The Mk-BW Zircaloy grid corner has been improved over the Mk-BZ design by replacing the corner keying window with a corner column which provides improved strength and reduced hang-up potential.

The Mark-BW fuel assembly is fully compatible with Westinghouse reactor internals and resident fuel and will perform well during fuel assembly handling and storage maneuvers. The exterior features of the Zircaloy intermediate spacer grid have been based on successful Zircaloy spacer grid designs (Reference 3). The mechanical compatibility and superior handling performance of the Mk-BW fuel assembly design have been demonstrated by an exercise with the prototype test fuel assembly at the Catawba, Unit 2 facilities and by the insertion of the four lead assemblies in McGuire, Unit 1.

4.2.1 Shipping and Handling Loads

The Mark-BW fuel assembly is designed to maintain its dimensional and structural integrity when subjected to normal handling and shipping loads. The lead-in tabs between the fuel rods on the upper and lower edges of the outer strips of the Mark-BZ design have been maintained in the Mark-BW design to provide better resistance to hangup during fuel handling. To preclude spacer grid failures resulting from excessive clamping loads imposed by the shipping container, B&W has developed a bracket that mechanically limits the imposed loads. The following provides the design bases and evaluation for handling and shipping loads.

Design Bases

Handling Loads

- Axial Pull : FA Wet Weight + [ ] Pounds
- Axial Push : [ ] Pounds

Shipping Loads

- Lateral : 6.0 G's load factor
- Axial : 4.0 G's load factor

4.2.1.1 Shipping Loads Evaluation

An analysis was performed to ensure the structural adequacy of the Mark-BW design under the above specified shipping loads. The results of the analysis showed that the fuel assembly and its components will maintain their structural integrity under the specified shipping loads. Crush tests as described in Section 3.3.5.2 have been performed to ensure that the spacer grid dimensional stability is maintained during normal shipping and handling conditions. Summaries of the results are provided next.

1. The upper and lower plenum springs are designed to maintain the fuel column position and prevent the formation of axial gaps. This is done by maintaining a preload on the fuel stack to counter acceleration loads up to '4G' which is significantly above those expected to occur during shipping and handling.

2. The fuel rod will not slip through the spacer grids under the maximum axial shipping loads.

3. The spacer grids will maintain their structural integrity under the maximum lateral shipping loads, and the maximum clamping loads.

4. Spacer grid soft stops are designed to maintain acceptable fuel rod grip forces under the ' 6G ' lateral shipping loads.

4.3 Materials Compatibility

The majority of the materials used in the design of the Mark-BW Fuel Assembly includes 304L Stainless Steel, Inconel 718, and Zircaloy-4 which surround and contain the uranium dioxide (UO_2) fuel pellets. A list of materials is shown in Table 4-6 with their respective use on the Mark-BW fuel assembly. Each material component is based on an industry-wide standards and modified according to individual engineering concerns. An example of this type of modification is lowering the cobalt content in stainless steel and inconel components to decrease the activity levels on non-fuel components in-reactor without affecting material performance (Reference 13). The following sections summarize the material and environmental compatibility for the Mark-BW fuel assembly components. The materials on Mark-BW fuel assemblies have been demonstrated through industry practice to be acceptable, and all current Mark-BW material specifications are found to be compatible with those used in the Seismic and Loss Of Coolant Accident (LOCA) analyses.

4.3.1 Component Material Properties

Some mechanical and metallurgical properties of the zircaloy, stainless steel, and inconel metals are very sensitive to the fabrication process. Changing the amounts of alloying elements, annealing temperatures, and percent of cold work in the metal can vary the strength and ductility of the final product. Through extensive research and development and industry-wide experience, manufacturing techniques are continually adjusted to improve particular material performances. The following paragraph describes the major accomplishments on material improvements to support the Mark-BW program.

Optimizing the alloy additions in Zircaloy-4 cladding combined with modified thermo-chemical treatments allows the recrystallized alloys to show better creep strength with minimal effects on the tensile strength and ductility (Reference 12). This advanced Zircaloy-4 cladding will be used on the Mark-BW fuel assemblies. Zircaloy-4 spacer grids use standard base material and exhibit excellent strength and ductility.

Stainless steels have been used throughout the nuclear industry, and wrought 304L SS used on the Mk-BW fuel assembly possess very good formability and can be readily welded by all common methods. All welding on the Mark-BW fuel assemblies consists of joining metals of similar composition (i.e. welding Zircaloy-4 to Zircaloy-4). Therefore, compatibility of mating components is acceptable from a mechanical and metallurgical standpoint.

Inconel 718 alloys have been markedly improved by increasing the annealing temperature and introducing more cold work into the final process. This results in a much improved uniform grain structure and a more ductile metal. The uniform grains, especially on the surface of the material, improve the metals resistance to defects that may cause cracking. Understanding and optimizing the material processes will improve the compatibility of the fuel assembly components to their mechanical and metallurgical effects.

4.3.2 Corrosion and Irradiation Effects

The excellent corrosion resistance of the metals in the Mark-BW fuel assemblies has been demonstrated by examination of production Mark-B fuel assemblies currently in-reactor. This corrosion resistance is the result of a combination of high quality base materials and sound manufacturing techniques. Material standards were specified to insure high quality base material and controlled manufacturing procedures were developed to produce components with a minimum of surface contamination. Manufacturing, handling, and welding procedures have also been developed to prevent surface contamination of components and to insure that contaminants do not come into contact with the metals during welding or annealing operations.

Zircaloy-4 is chemically reactive and readily forms various oxide compounds when subjected to temperatures characteristic of the reactor environment. The corrosion resistance of Zircaloy-4 is sensitive to the composition of the base material and any organic surface contamination. When subjected to the reactor environment, a thin, tough, adherent oxide film forms and acts as a protective barrier against corrosive attack. Rigorous inspections and tests compliment the manufacturing process to assure a corrosion resistant material.

The lower carbon in 304L SS provides excellent resistance to intergranular corrosion and limits sensitization of the metal. The Zircaloy-4, 304L SS, and Inconel 718 components retain their high impact strength after irradiation, and overall strength is increased while the ductility of the metal is reduced when in service. The activity levels caused by non-fuel components in-reactor is minimized by reducing the level of cobalt in 304L stainless steel and Inconel 718 components on e Mark-BW fuel assemblies.

The use of Zircaloy-4 for tubing and strip applications, 304L SS for plate and barstock needs, and Inconel 718 for strip and holddown spring applications are compatible from a corrosion standpoint on Mark-BW fuel assemblies.

4.3.3 Summary

The component materials used on the Mark-BW fuel assembly are compatible with the reactor environment and with each other.

4.4. Operational Experience

The Mark-BW fuel assembly design is based on irradiation experience obtained by both B&W and NFI. The irradiation experience extends to burnups of 52,000 MWd/mtU, and comprises six fuel assembly designs in five different type reactor plant designs. Summaries of irradiation experience of B&W and NFI are presented in the following sections. Planned poolside measurements to be performed on the Mark-BW LAs are discussed.

4.4.1. B&W 177 Plant (Mark-B) Fuel Experience

As of December 31, 1987 a total of 967,888 B&W manufactured zircaloy clad fuel rods have been irradiated. In Table 4-7 are the maximum incore and discharged fuel assembly burnups for each of eight BWFC supplied 177 reactor plants. Table 4-8 lists the burnup range for all fuel assemblies irradiated in these same reactor plants.

4.4.2. NFI Fuel Experience

Table 4-9 lists the irradiation experience for NFI supplied fuel assemblies.

4.4.3. B&W Mark-BW Lead Assembly

The Mark-BW LAs are being irradiated in McGuire Unit 1 in cycles 5, 6 and 7. The fuel assemblies will be inspected poolside following each cycle of exposure. The following inspections will be performed:

1. Visual (Video)
2. Fuel Assembly Growth.
3. Shoulder Gap (On selected peripheral rods)
4. Holddown Spring Set.
5. Fuel Rod Growth (On selected peripheral rods)
6. Spacer Grid Position.
7. Oxide Thickness.
8. Rod Diameter.

The Mark-BW LAs were characterized during manufacture and the data recorded. This will allow for more precise calculations when comparing PIE data with the as-built data.

4.5 Fuel Rod Mechanical Evaluation

A series of analyses have been performed on the Mark-BW fuel rod design to confirm its in-reactor mechanical performance. The areas that were analyzed include:

A. Cladding Stress

B. Cladding Strain

C. Cladding Fatigue

D. Creep Collapse

E. Fuel rod growth

G. Shipping and handling

Items E and G are discussed in the fuel assembly design evaluation section 4.1. Items A through D are discussed after the following section on the impact of oxide formation of the analyses:

4.5.1 Oxide Growth

The growth of an oxide layer on the cladding has several effects. The effective thickness of the cladding is reduced as the base metal is converted to oxide. The cladding operates at a higher temperature because of the lower thermal conductance of the oxide compared to the base metal. In addition, hydrogen released from the Zirc-water reaction will diffuse into the cladding forming Zirc-hydrides. The presence of Zirc-hydrides can result in a decrease in cladding ductility.

The Mark-B oxide experience demonstrates that the maximum span value expected at high burnup for the Mark-BW is [ ]. This value was used as the maximum oxide layer thickness for the Mark-BW analyses where required.

The creep collapse analysis is based on the cladding ovalizing into a gap in the fuel column. If the gap were not present the cladding would come into contact with the pellets which would slow or stop further ovalization. The cladding temperature in the region of the gap is significantly lower, only a few degrees above coolant temperature for both the inside and outside of the cladding. At these lower temperatures the cladding is oxidized at a much lower rate. Because of this, much thinner layers of oxide are expected on the cladding in the region of a gap. The conservatism in the creep collapse analysis is sufficient to account for oxide layer growth (Reference 26).

In the strain analysis the conservatism in the TACO2 code is sufficient to account for the effects of oxide layer growth (Reference 26). The cladding hydrogen content was determined assuming an oxide layer of [ ] mils thickness and a [ ] hydrogen pickup fraction which is conservative for high burnup fuel. The cladding hydrogen content was found to be [ ] ppm. Cladding embrittlement requires concentrations of around [ ] ppm (Reference 26). The oxide growth that the Mark-BW fuel rod design is projected to experience will not result in embrittlement of the cladding.

In the fatigue analysis the cladding thickness is reduced to account for the effects of oxide layer growth and cladding dimensional tolerances.

4.5.2 Fuel Rod Cladding Stress

The fuel rod cladding is analyzed for the stresses induced during operation. The ASME pressure vessel stress intensity limits are used as guidelines. Conservative values are used for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, differential temperature and unirradiated cladding yield strength. The analysis results for the Mark-BW LTAs show that the maximum cladding stress intensities are within limits under all Condition I and II events.

Pellet-cladding interaction (PCI) and creep collapse induced stresses are not of concern as small deformations of the cladding will relieve those stresses. Limits are based on ASME criteria. 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 Sm. Sm is equal to 2/3 of the minimum specified unirradiated yield strength of the material at the operating temperature [ ] deg F). The limits are as follows:

I. Primary general membrane stress intensities (Pm) must not exceed Sm.

II. Local primary membrane stress intensities (Pl) must not exceed 1.5 * Sm. These include the contact stresses from spacer grid-fuel rod contact. Primary membrane + Bending stress intensities (Pl +Pb) must not exceed 1.5 * Sm.

III. Primary membrane + Bending + Secondary stress intensities (Pl+Pb+Q) must not exceed 3.0 * Sm.

The fuel rod stresses under faulted conditions are evaluated using primarily the methods outlined in Appendix F of the ASME Boiler and Pressure Vessel Code, Section III. The results of analyses under faulted conditions are reported in section 4.1.2.

Stress intensity calculations combine the stresses so that the stress intensity is maximized.

Pressure and temperature inputs to the stress intensity analyses are chosen so that the operating conditions for all Condition II transients are enveloped.

The type of stresses which are analyzed in this calculation are as follows:

1. Pressure Stresses - These are membrane stresses due to the external and internal pressure on the fuel rod cladding.
2. Flow Induced Vibration - These are longitudinal bending stresses due to vibration of the fuel rod. The vibration is caused by coolant flow around the fuel rod.
3. Ovality - These are bending stresses due to external and internal pressure on the fuel rod cladding that is oval. This does not include the stresses resulting from creep ovalization into an axial gap.
4. Thermal Stresses - These are secondary stresses that arise from the temperature gradient across the fuel rod during reactor operation.
5. Fuel Rod Growth Stresses - These secondary stresses are due to the fuel rod slipping through the spacer grids. These may be due to the fuel assembly expanding more than the fuel rod due to heatup, or they may be due to fuel rod growth from irradiation.
6. Three point Grid Stop Stresses - These are bending stresses due to the grid stop loads against the fuel rod cladding.
7. Fuel Rod Spacer Grid Interaction - These are localized stresses due to contact between the fuel rod cladding and the spacer grid stops.

Classifications of Stresses:

Loading Condition	Stress Category
Pressure Stresses	P_m
Ovality Stresses	P_b
Spacer Grid Interaction	P_l
Flow Induced Vibration	P_b
Radial Thermal Expansion	Q
Differential Rod Growth	Q

P_m, primary membrane stresses

P_b, primary membrane bending stresses

P_l, primary membrane local stresses

Q, secondary stresses

The results of the stress analysis for the Mark-BW are:

Loading Condition	Stress Intensity Limit	Minimum Margin %
Primary Membrane	S_m	[ ]
Primary Membrane + Bending	1.5 S_m	[ ]
Primary Membrane + Bending + Secondary	3.0 S_m	[ ]

Margins are calculated by the following:

Margin % = [(Allowable - Predicted)/Predicted] x 100 %

The minimum unirradiated yield strength of the cladding used is [ ] psi.

The Mark-BW design will operate with sufficient margin for cladding stress.

4.5.3. Fuel Rod Cladding Strain

The fuel rod is analyzed to determine the maximum transient the fuel rod could experience before the transient strain limit of [ ] (Reference 4) was exceeded. The transient strain limit uses cladding circumference changes before and after a linear heat rate (LHR) transient to determine the strain.

The strain analysis for the Mark-BW rod is based on the method used for the Mark-B fuel assembly design given in reference 23. Some changes are required due to the design differences between the Mark-BW design and the Mark-B design. The smaller pellet to cladding gap for the Mark-BW design as opposed to the Mark-B design results in significantly higher value for linear heat rate to melt (LHRM). Since the fuel will exceed the strain criteria before the LHRM value is reached, the linear heat rate (LHR) at which the strain criteria is exceeded will set transient strain criteria. For the transient strain calculation using TACO2 the fuel rod is ramped at [ ] MWd/mtU from [ ] KW/ft LHR in steps until the 1% strain criteria is exceeded. The strain is based on changes in the fuel pellet diameter over the diameter at [ ] KW/ft.

The formula for determining the transient strain is:

$$E_{TRANS} = \frac{\text{(Pellet O.D.) Transient} - \text{(Pellet O.D.)}_o}{\text{(Pellet O.D.)}_o} \times 100\% \leq [\quad]$$

In the analysis for the Mark-BW fuel rod the transient increase required to exceed the 1% limit was found to be [ ] KW/ft, which is much greater than the maximum transient the fuel rod is expected to experience. This represents a peak LHR of [ ] KW/ft (Transient from [ ] KW/ft).

4.5.4. Fuel Rod Cladding Fatigue

The fuel rod was analyzed for the total fatigue usage factor using the ASME pressure vessel code as a guideline. A maximum fatigue usage factor of [ ] is allowed. The O'Donnell-Langer fatigue curve for irradiated Zircaloy was used, and a fuel rod life of [ ] years was assumed. A vessel life of [ ] years was assumed. The fuel rod cladding will experience [ ] or [ ] of the number of transients the reactor pressure vessel will experience. All possible condition I & II events expected and one condition III event are analyzed to determine the total fatigue usage factor experienced by the fuel rod cladding. Conservative inputs in terms of cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure and differential temperature across the cladding are assumed. A summary of the system transients considered is listed in Table 4-10. The combination of transients, and the power histories that are analyses to determine the fatigue usage factor are given in Table 4-11.

The results of the fatigue analysis for the Mark-BW show a maximum fatigue usage factor of [ ] which is well under the limit of [ ]

4.5.5 Fuel Rod Cladding Creep Collapse

The fuel rods are analyzed for creep collapse using methods outlined in reference 23 and reference 25. These are the topicals for the fuel rod thermal code TACO2 and the creep collapse code CROV. The acceptance criteria is that the predicted creep collapse life of the fuel rod must exceed the maximum expected incore life. CROV predicts that the fuel rod will fail due to creep collapse when either of the following happens:

1. The rate of creep ovalization exceeds [ ] mils/hr.
2. The Maximum fiber stress exceeds the unirradiated yield strength of the cladding.

The following conservatisms are used in determining creep collapse life of the fuel rod:

1. Minimum fuel rod pre-pressure is used.
2. No fission gas release is assumed.
3. Worst case densification is used.
   a. [ ]% TD change at [ ] MWd/mtU for [ ]% TD fuel.
   b. [ ]% TD change at [ ] MWd/mtU for [ ]% TD fuel.
4. A factor of [ ] is used on Power history or burnup to account for uncertainty.
5. A worst case or enveloping power history is used.
6. Worst case cladding dimensions are used.
   a. [ ] for cladding thickness.
   b. [ ] for cladding ovality.

These conservatisms are used in determining the creep collapse life of the Mark-BW fuel rod. The same method will be used in determining the creep collapse life of all Mark-BW reload fuel.

The creep equations used in CROV are based on measurements performed on the Mark-B cladding to determine the thermal creep rate, and the cladding texture. Those measurements and post irradiation examination (PIE) data were used to derive the irradiation creep rate. Thermal creep and texture measurements were made on the cladding used for the Mark-BW IAs to verify that the texture and creep were similar to those for Mark-B. It was assumed that the irradiation creep component will be the same as for the Mark-B. This will be verified from PIE data gathered poolside on the Mark-BW IAs.

Using nuclear design inputs, a power history was determined for the Mark-BW fuel rod. This power history with appropriate uncertainty factors was input into the computer code TACO2 which determined the temperature, the pressure, and the fast neutron flux level history of the Mark-BW fuel rods. This history was input in CROV using conservative cladding dimensions. From the output of CROV the creep collapse point of the Mark-BW fuel rods was determined. This was [ ] MWd/mtU burnup and an incore exposure of [ ] EFPH.

Table 4-1. Limiting Load Conditions for Fuel Assembly Components for Normal Opertation

Component	Load Condition	Basis for Design Limit	Design Limit	Actual Load	% Margin
Guide Thimble					
Upper Nozzle					
Lower Nozzle					
Holddown Spring					

Table 4-1. Limiting Load Conditions for Fuel Assembly Components for Normal Opertation (Cont'd)

Component	Load Condition	Basis for Design Limit	Design Limit	Actual Load	% Margin
Holddown Spring					
Instrument Thimble					

Table 4-2. Seismic and LOCA Induced Maximum Spacer Grid Impact Forces

Grid Impact Loading Case	Maximum Grid Impact Force, lbs.
McGuire LOCA	
Catawba LOCA	
McGuire SSE	
Catawba SSE	

Differences due to Catawba configuration as a upflow plant, and McGuire configuration as a downflow plant.

Table 4-3. Component Vertical LOCA Forces

Component	LOCA Case	Maximum Load (lbs)
Guide Thimble (Load/Thimble) Limiting	CLB - EOL	
Fuel Rod (Load/Rod)	CLB - EOL	
Lower Nozzle	CLB - EOL	
Holddown Spring	CLB - EOL	

Table 4-4. Mark-BW Fuel Assembly Stress Analysis Results for SSE and Combined SSE Plus LOCA Conditions

Component	
Guide(b) Thimble, lbs	
Fuel Rod,(b) ksi	
Upper Nozzle ksi	
Lower Nozzle ksi	
Spacer Grid	

Table 4-5. Summary of Reactor Coolant System Design Transients

Normal Conditions	Occurrences	Temperature Range °F
1. Heatup and cooldown at 100°F/hr (pressurizer cooldown 200°F/hr)		
2. Unit loading and unloading at 5% of full power/min.		
3. Step load increase and decrease of 10% of full power		
4. Large step load decrease,		
5. Steady state fluctuations		
Upset Conditions		
1. Loss of load, without immediate turbine or reactor trip		
2. Loss of power (blackout with natural circulation in the Reactor Coolant System)		
3. Loss of flow (partial loss of flow one pump only)		
4. Reactor trip from full power		
5. Inadvertent auxiliary spray		
6. Operating Basis Earthquake		
Steam Generator, Reactor Coolant Pump, Pressurizer		
Reactor Vessel, Unit 1		
Reactor Vessel, Unit 2		

Table 4-6. Mark-BW Fuel Assembly Materials

Material	Reference Standard	Fuel Assembly Component Parts

Table 4-7

Operating Status of B&W Fuel and Designs Reactors

(December 31, 1987)

Reactor	Reactor Cycle	Incore	Maximum Assembly Burnup, MWd/mtU Discharged to Date

Table 4-8

Summary of Burnup Experience for B&W Supplied Zircaloy Clad Fuel

(December 31, 1987)

Fuel Assembly Average Burnup MWd/mtU	Assemblies Incore on Dec. 31, 1987 Number Assy's	Rods	Assemblies Discharged Through Dec. 31, 1987 Number Assy's	Rods

Table 4-9

NFI Fuel Irradiation Experience

(September 30, 1987)

Reactor	Fuel Type	Number of Fuel Assemblies (Total)	Beginning of Irradiation	Max Assembly Burnup (MWd/mtU)

Table 4-10

Summary of Reactor Coolant System Design Transients

Normal Conditions	Occurrences
1. Heatup and cooldown at 100 F/hr (pressurizer cooldown 200 F/hr)	
2. Unit loading and unloading at 5% of full power/min.	
3. Step load increase and decrease of 10% of full power (FP).	
4. Large step load decrease, (95% of full power with steam dump.	
5. Steady-state fluctuations	
Upset Conditions	
6. Loss of load, without immediate turbine or reactor trip.	
7. Loss of power (blackout with natural circulation in the reactor coolant system)	
8. Loss of flow (partial loss of flow one pump only	
9. Reactor trip from full power	
10. Inadvertent auxiliary spray	
Condition III Event	
11. Minor loss of coolant accident or secondary steam line break	

Table 4-11

Transient Group Core Power History

Group Number (transient) from above	Number of Fatigue Cycles	Core Power History (% FP)

Terms: HZP - Hot Zero Power

FP - Full Power

FIGURE 4-1

HORIZONTAL CORE SEISMIC AND LOCA MODEL

FIGURE 4-2

FA VERTICAL MODEL

FIGURE 4-3

Total Force on Core During Hot Leg Break
(Guillotine at Nozzle of 14-inch I.D. Surge Line-
Attachment to Hot Leg)

FIGURE 4-4

Total Force on Core During Cold Leg Break (Guillotine at Nozzle of 10-Inch I.D. Safety Injection Line-Attachment to Cold Leg)

Figure 4-5. Mark BW Grid Impact Force Responses - Outer Peripheral Fuel Assembly Case McGuire LOCA

5. Mark-BW Thermal-Hydraulic Design Evaluation

5.1 Introduction

This section describes the thermal-hydraulic characteristics and analysis methods to be used in the licensing evaluation of the Mark BW fuel assembly.

The basis for the thermal-hydraulic design of the fuel assembly is to enable the reactor to operate at rated power with sufficient margin to withstand operational and moderate frequency transients without sustaining damage to the core. To insure core integrity the following design criteria have been established:

1. The minimum departure from nucleate boiling (DNB) ratio shall be sufficiently high to provide a 95% probability, with 95% confidence that no fuel rod will experience departure from nucleate boiling during normal operation or incidents of moderate frequency.

2. No fuel melting will occur at design conditions, including design overpower.

3. The fuel assembly will not be permitted to lift off the lower core plate during normal operating conditions.

The following section describes the analyses and evaluations which have been or will be performed to assure that these criteria are satisfied by the Mark-BW fuel assembly. Where the thermal-hydraulic analyses described in this section have not been completed for the final production assembly, the procedures described here will be followed.

5.2 Core Pressure Drop

As described in Section 3.3, the pressure drop characteristics of the Mark-BW fuel assembly were determined through a series of flow tests at the Control Rod Drive Line (CRDL) at B&W's Alliance Research Center. The results of these tests were used as the basis for the calculation of formloss coefficients for the end fittings and spacer grids.

The fundamental equation for determining pressure drop across a span is defined as follows:

$$\text{Unrecoverable Pressure Drop} = (\text{delta } P_f + \text{delta } P_{c/e})$$

where delta P_f = friction pressure drop, psi
delta $P_{c/e}$ = contraction/expansion pressure drop, psi

The friction and contraction/expansion losses which make up the total unrecoverable pressure drop are dependent on the velocity head and the component loss characteristics. The following relationships define the two components of the total pressure drop loss.

$$\text{delta } P_f = \frac{fL}{D_e} \left[\frac{\text{rho}(V^2)}{2g_c}\right]$$

and

$$\text{delta } P_{c/e} = k_{lift} \left[\frac{\text{rho}(V^2)}{2g_c}\right]$$

where
delta P_f = friction pressure drop, psi
f = friction factor
D_e = hydraulic diameter in tubed region, ft
rho = coolant density, lb_m/ft^3
g_c = constant, 32.174 $ft\text{-}lb_m/lb_f\text{-}ft^2$
delta $P_{c/e}$ = cont/exp pressure drop, psi
k_{lift} = fuel assembly lift coefficient
V = coolant velocity, ft/sec
L = length of fuel rods, ft

In order to determine the formloss coefficient across a span the total loss due to friction was subtracted from the total pressure loss across the span. For the CRDL tests the total friction loss was determined using two methods. The first method was based on the measured friction loss across the span and led to the following equation:

$$k = \frac{2gA_B^2}{j_B w_B^2} \left(\text{delta } P - \text{delta } P_f \right) \left[\frac{rho_{cal}(g)}{g_c} \right] + \frac{A_B^2}{w_B^2} \left[\frac{W_1^2}{A_1^2} - \frac{W_2^2}{A_2^2} \right]$$

where j_B = coolant weight, lb_f/ft^2, $rho(g)/g_c$
g = gravitational acceleration, 32.174 ft/sec^2
g_c = 32.174 lb_m-ft/lb_f-sec^2
A_1, A_2 = flow areas at Points 1 and 2, ft^2
W_1, W_2 = flow rates at Points 1 and 2, ft^3/sec
A_B = fuel bundle tubed region flow area, ft^2
w_B = fuel bundle tubed region flow rate, ft^3/sec
rho_{cal} = calibration/measurement density 62.43 lb_m/ft^3
delta P = measured pressure loss, ft
delta P_f = measured friction loss, ft
k = formloss coefficient

The second method was based on the calculated friction loss across the span. Using this method the friction loss was based on the Reynolds Number relationship $f = .186(Re)^{-0.2}$ which led to the following equation:

$$k = \frac{2gA_B^2}{j_B w_B^2} \left(\text{delta } P \right) \left[\frac{rho_{cal}(g)}{g_c} \right] - \frac{fL}{D_e} + \frac{A_B^2}{w_B^2} \left[\frac{W_1^2}{A_1^2} - \frac{W_2^2}{A_2^2} \right]$$

where f = friction factor
L = tubed length region in span, ft
D_e = hydraulic diameter in tubed region, ft

The equation based on Reynolds Number yielded slightly higher, more conservative formloss coefficients than the equation based on measured friction loss. Therefor the formloss coefficient based on Reynolds number was chosen. Using the results from the CRDL tests, the formloss coefficients over the entire range of flow conditions tested were determined using the equations above. Using these results, a curve fit was performed for each fuel assembly component to provide a relationship in the form $k = A(Re)^b$ to calculate the formloss coefficient as a function of Reynolds Number only.

In order to determine the pressure drop characteristics of a transition core into which the Mark-BW is being placed, some knowledge of the pressure drop characteristics of the resident Westinghouse fuel is required. For the analysis of the Mark BW lead assemblies (LA's), in Duke Power Company's McGuire 1 core, B&W was able to determine formloss coefficients of the Westinghouse OFA analytically from core pressure drop data.

Analyses have been performed using the LYNX1 and LYNX2 codes (references 18 & 19) to establish pressure drop characteristics of the Mark-BW in a Westinghouse OFA core. These analyses compared the overall pressure drop of the Mark BW and OFA assemblies, as well as the pressure drop of individual components. It was determined that the pressure drop of the first intermediate spacer grid was approximately [ ] on the Mark-BW than on the OFA. This was due to the Mark-BW's having a non-mixing grid while the OFA had a mixing grid. All other Mark-BW grid pressure drops were within [ ] of the corresponding OFA values. In addition, the Mark-BW has a 6% smaller flow area than the OFA. The combined effect of these differences resulted in a diversion of flow from the OFA's to the Mark-BW's below the core midplane and from the Mark-BW's to the OFA's above the core midplane. Even with this flow diversion, the amount of crossflow was found to be less than the [ ] maximum crossflow criteria. Using the LYNX1 code and the component formloss coefficients, the overall core pressure drop for a full Mark-BW core with the standard bottom nozzle (Figure 3-5) was calculated to be [ ] psi. This value was generated at the same conditions as those that produced a [ ] psi core pressure drop for a full OFA core.

5.3 Fuel Assembly Hydraulic Lift

The hydraulic lift force on a fuel assembly is attributed to the unrecoverable pressure drop across the length of the assembly. As part of the pressure drop testing performed on the Mark BW fuel assembly at the Alliance Research Center's Control Rod Drive Line, a series of hydraulic lift tests were performed. A brief description of the testing procedure was provided in Section 3.3. Results from these tests were used to develop assembly and component hydraulic lift formloss coefficients. It should be noted that for all components except the top nozzle, the hydraulic lift formloss coefficients are equal to the pressure drop formloss coefficients which were determined from pressure drop measurements as described in Section 5.2. Since the coolant flow is not fully developed immediately upon exiting the fuel assembly, the recoverable component of the top nozzle must be considered as an unrecoverable component, thereby increasing the formloss coefficient for the upper end fitting. Using the hydraulic lift test data, an expression for the overall hydraulic lift formloss coefficient for the assembly as a function of Reynolds Number was determined. Knowing this expression and the corresponding pressure drop formloss coefficients of the individual assembly components, the top nozzle hydraulic lift formloss coefficient was determined. The resulting component lift formloss coefficients were then input into LYNX1. Results from the LYNX1 analysis showed that the pressure drop which was predicted with the calculated formloss coefficients was within 0.9% of the measured test data.

Using the calculated formloss coefficients and the LYNX1 code, along with bounding assumptions on flow conditions (i.e. maximum flow) and core configuration, several analyses were performed which evaluated the hydraulic lift forces on the Mark BW in both a full BW core environment and in a mixed core environment. The mixed core analysis showed that the total lift force for the Mark-BW would be [ ] in a full Mark-BW core environment than in a Westinghouse OFA environment. An evaluation of isothermal operation at zero power and operation at full power showed that the point of minimum lift margin occurred with operation at power. This indicates that there is no need for designating a fourth pump start-up temperature for the Mark-BW fuel assembly, as is commonly done for B&W Mark-B fuel.

5.4 Core DNB Analysis

The purpose of the core DNB analysis is to insure that there is 95% probability, with a 95% confidence level that no fuel rod will experience a departure from nucleate boiling (DNB) during normal operation or transients of moderate frequency. Current thermal-hydraulic analysis methodology employs the LYNX series of crossflow analysis codes (LYNX1, LYNX2, LYNXT, references 18, 19, and 20) to calculate subchannel departure from nucleate boiling ratios (DNBR's). The DNB criterion is met if the calculated DNBR is greater than the design limit DNBR, which has been established for the critical heat flux (CHF) correlation being used. For the Mark-BW DNB analysis, a one-pass LYNXT model will be employed using the statistical core design (SCD) analysis technique developed in reference 21. The BWCMV CHF correlation (developed in reference 22) will be employed in the SCD analysis.

The advantage of the SCD analysis technique is that it treats core state and bundle uncertainties statistically. Traditional analysis methods assume the worst level of each uncertainty occurs simultaneously. As stated in reference 21, in the SCD method, input uncertainties are analyzed using statistical methods and an overall DNBR uncertainty is determined. This is then used to establish a design limit DNBR known as the Statistical Design Limit (SDL). All variables treated in the development of the SDL are then input into the thermal-hydraulic analysis computer codes at their nominal values. All other variables continue to be input at conservative levels. Once the SDL has been established, the calculated DNBR, at a specific core state, is compared to the SDL to determine if the DNB protection criterion is met. For plant specific applications, margin will be added to the SDL to define an analysis limit known as the Thermal Design Limit (TDL). Using the following formula, the level of retained thermal margin made available by the use of the TDL can be calculated.

$$\text{Retained Thermal Margin (\%)} = \frac{TDL - SDL}{TDL} \times 100$$

This retained thermal margin is then used to provide flexibility in the design of fuel cycles for reload cores. Examples of offsets that might be assessed against the retained margin include transition core effects, and penalties for input uncertainties greater than those considered in the SDL development.

The SCD methodology will be implemented for all steady-state and transient DNB analyses performed for the Mark-BW fuel assembly. In general, core conditions for steady-state plant operation are bounded by the reactor core safety limits. These limits, which are generated using design radial and axial power distributions, graphically represent various combinations of pressure, temperature, and power that produce acceptable operation based on the thermal design and hot leg boiling limits (See Figure 5-1). To insure that the DNBR margin to the design limit DNBR determined using the design power distributions is preserved for other power distributions, maximum allowable peaking (MAP) limits are established. The MAP limits are a set of curves which represent, for a given core state point, various combinations of radial and axial peak and axial peak location that yield the same calculated DNBR as that calculated with design radial and axial peaking distributions. For each core analyzed, a full set of MAP limits is established and provided as input for the core maneuvering analysis. A set of typical MAP limits is presented on Figure 5-2.

The transient DNB analysis insures that the 95/95 DNB criteria is met. Several moderate frequency loss of coolant and overpower transients will be analyzed in order to determine the most limiting in terms of DNBR. This limiting DNBR will then be compared to the thermal design limit to insure that the DNB protection criteria is met.

5.5 Fuel Rod Performance

The fuel thermal performance is determined as a function of burnup using the TACO2 computer code (reference 23). The code includes calculational models for fuel densification and swelling, fuel restructuring, gas release, cladding creep and gap closure. Using these models, the TACO2 code conservatively calculates fuel pin temperature and pressure.

Figures 5-3 through 5-5 provide results from the fuel thermal analysis of the 96 percent theoretical density rod used in the Mark BW. Figure 5-3 shows the power history assumed for the TACO2 analysis, while Figures 5-4 and 5-5 provide fuel temperature and internal pressure results, respectively. The linear heat rate limit to prevent fuel centerline melt was calculated by TACO2 to be [ ] kw/ft. In general, application of these results to any other specific fuel cycle design, would require verification that the base inputs (densification, rod geometry, etc.) and power history used in the analysis bound those of the new design.

As an alternative to the TACO2 analysis, similar analyses may also be performed with the best estimate fuel thermal performance code TACO3 (reference 24). This code includes models for gap conductance, fuel densification and swelling, fuel restructuring, cladding creep and deformation, gap closure, and fission gas release. TACO3 provides a more accurate, less conservative fuel performance prediction than TACO2.

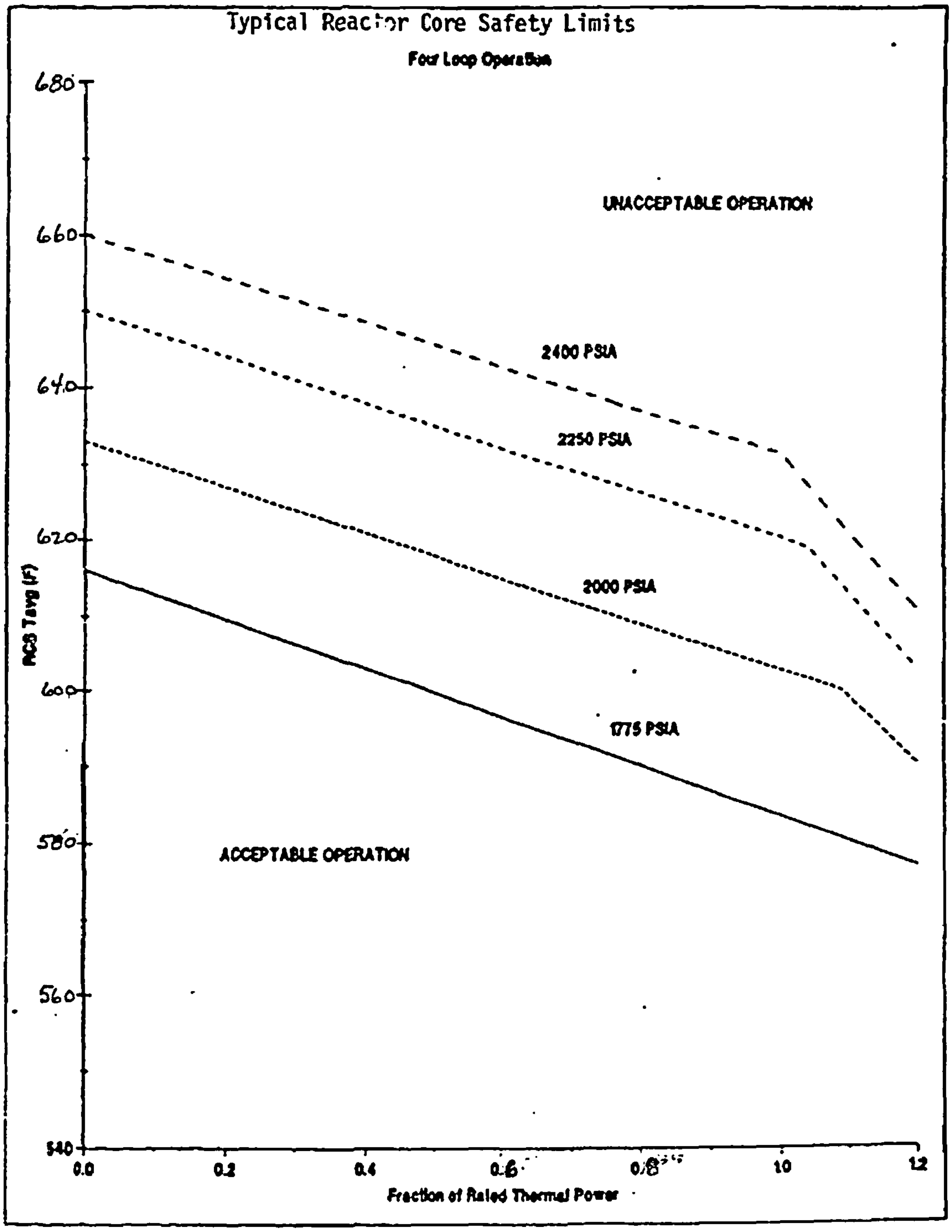
5.6 Transition Cycles

The thermal-hydraulic analysis of a Mark-BW transition core will be performed on a cycle-by-cycle basis. The purpose of the transition core analysis is to determine the effects of the transition core on various areas including core pressure drop, fuel assembly lift, calculated DNBR, and diversion crossflow. The goal of the analysis is to demonstrate compatible performance of the Mark-BW fuel with resident fuel.

One characteristic of a transition core which must be evaluated in terms of its effect on the core thermal-hydraulic analyses is the difference in pressure drop between the old and new fuel. By knowing which assembly has the lower pressure drop, one can determine in which direction flow diversion occurs, and, in turn, the effect of the increased or decreased flow on fuel assembly lift and calculated DNBR. For the Mark-BW lead assemblies (IA's) in Duke Power's McGuire 1 OFA core, the variation in core pressure drop along the length of the assembly caused flow to be diverted from the OFA's to the IA's below the core midplane and from the IA's to the OFA's above the core midplane.

One of the most critical areas of the transition core analysis is determining the impact of the transition core on calculated DNBR. The first step in analyzing this effect is to establish models of a full Mark BW core in both the single pass LYNXT code (reference 20) and the multipass LYNX1/LYNX2 combination (references 18 and 19). The next step is to analyze the actual mixed core configuration using the LYNX1/LYNX2 combination. If the mixed core analysis is more limiting than the full core analysis, then a transition core penalty is established. As discussed in Section 5.4 Core DNB Analysis, by using the Statistical Core Design technique, any transition core penalty will be offset against the retained margin made available by using the Thermal Design Limit.

FIGURE 5-1


Typical Reactor Core Safety Limits
Four Loop Operation
UNACCEPTABLE OPERATION
2400 PSIA
2250 PSIA
2000 PSIA
1775 PSIA
ACCEPTABLE OPERATION
RCS Tavg (F)
680
660
640
620
600
580
560
540
0.0
0.2
0.4
0.6
0.8
1.0
1.2
Fraction of Rated Thermal Power

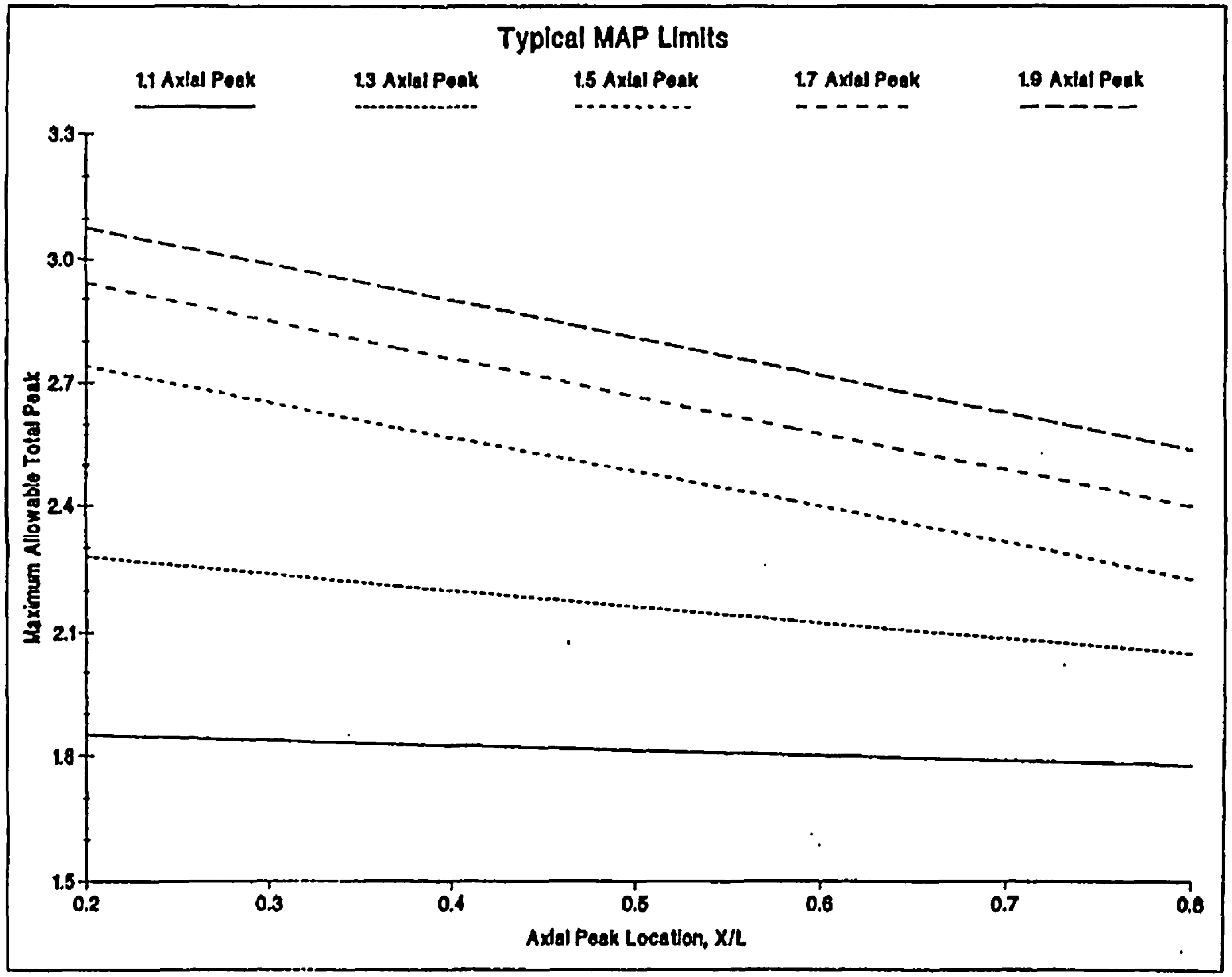

Typical MAP Limits
1.1 Axial Peak
1.3 Axial Peak
1.5 Axial Peak
1.7 Axial Peak
1.9 Axial Peak
Maximum Allowable Total Peak
3.3
3.0
2.7
2.4
2.1
1.8
1.5
0.2
0.3
0.4
0.5
0.6
0.7
0.8
Axial Peak Location, X/L

FIGURE 5-2

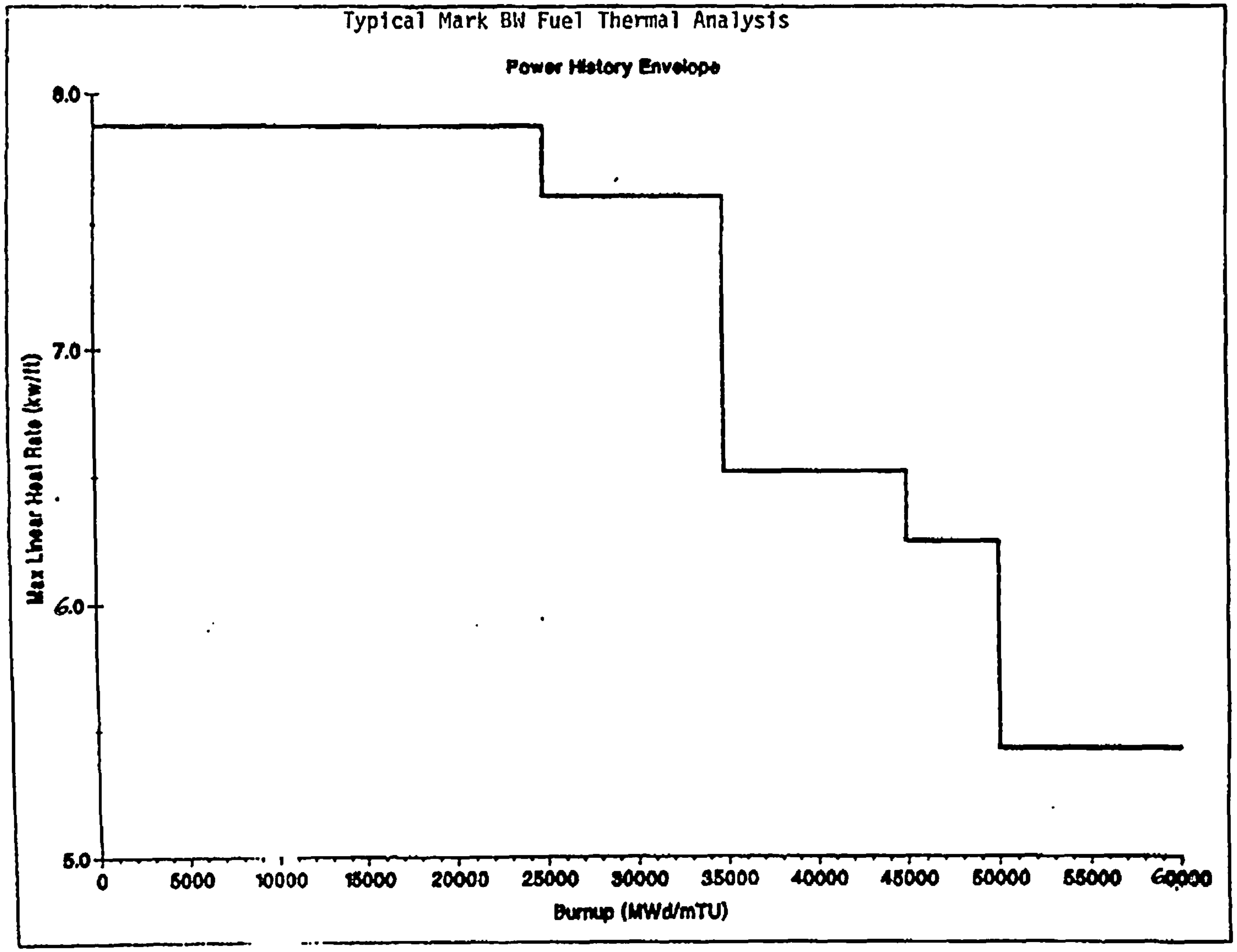
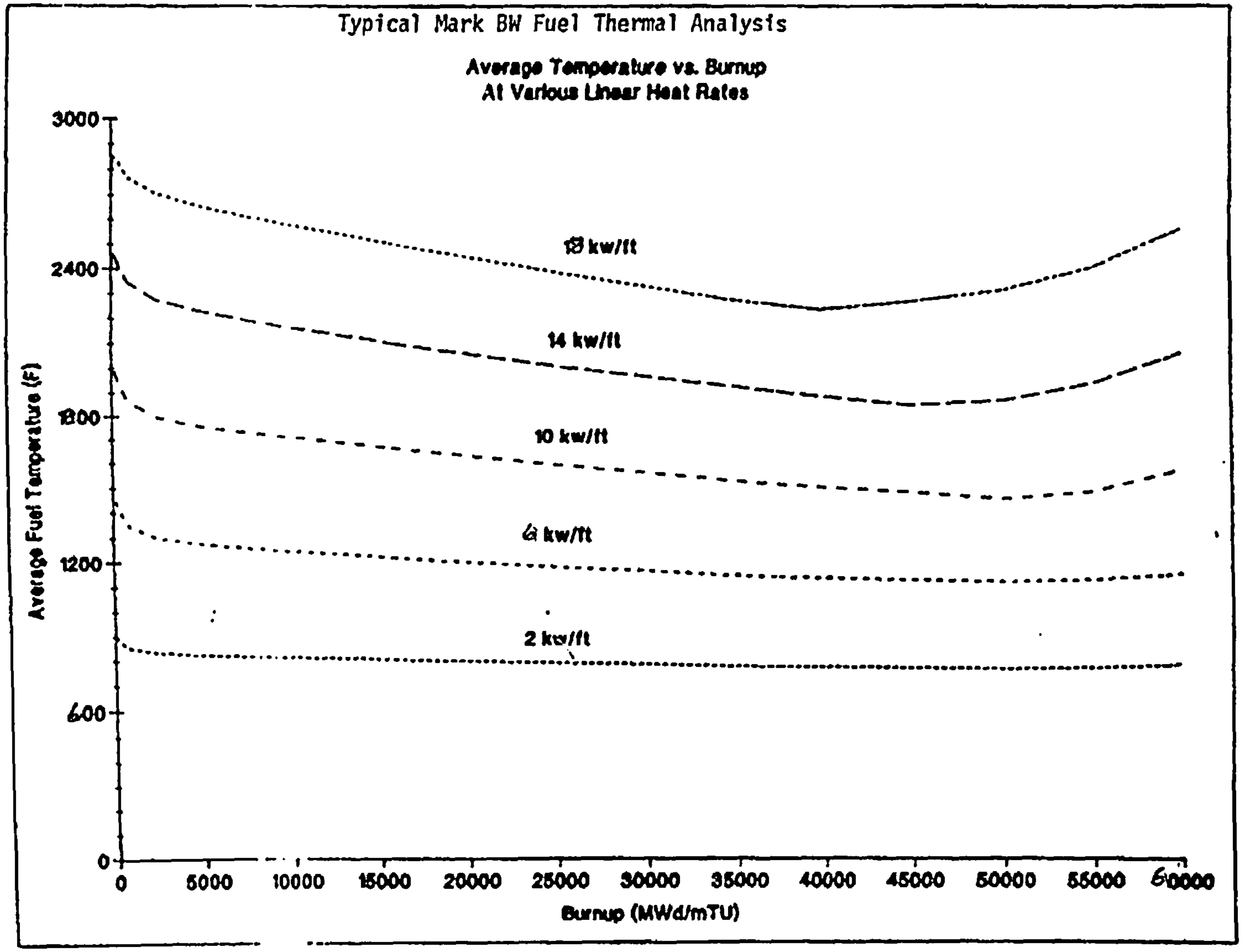

Typical Mark BW Fuel Thermal Analysis
Power History Envelope
Max Linear Heat Rate (kw/ft)
8.0
7.0
6.0
5.0
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
50000
55000
60000
Burnup (MWd/mTU)

FIGURE 5-3



Typical Mark BW Fuel Thermal Analysis
Average Temperature vs. Burnup
At Various Linear Heat Rates
18 kw/ft
14 kw/ft
10 kw/ft
6 kw/ft
2 kw/ft
3000
2400
1800
1200
600
0
Average Fuel Temperature (F)
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
50000
55000
60000
Burnup (MWd/mTU)

FIGURE 5-4

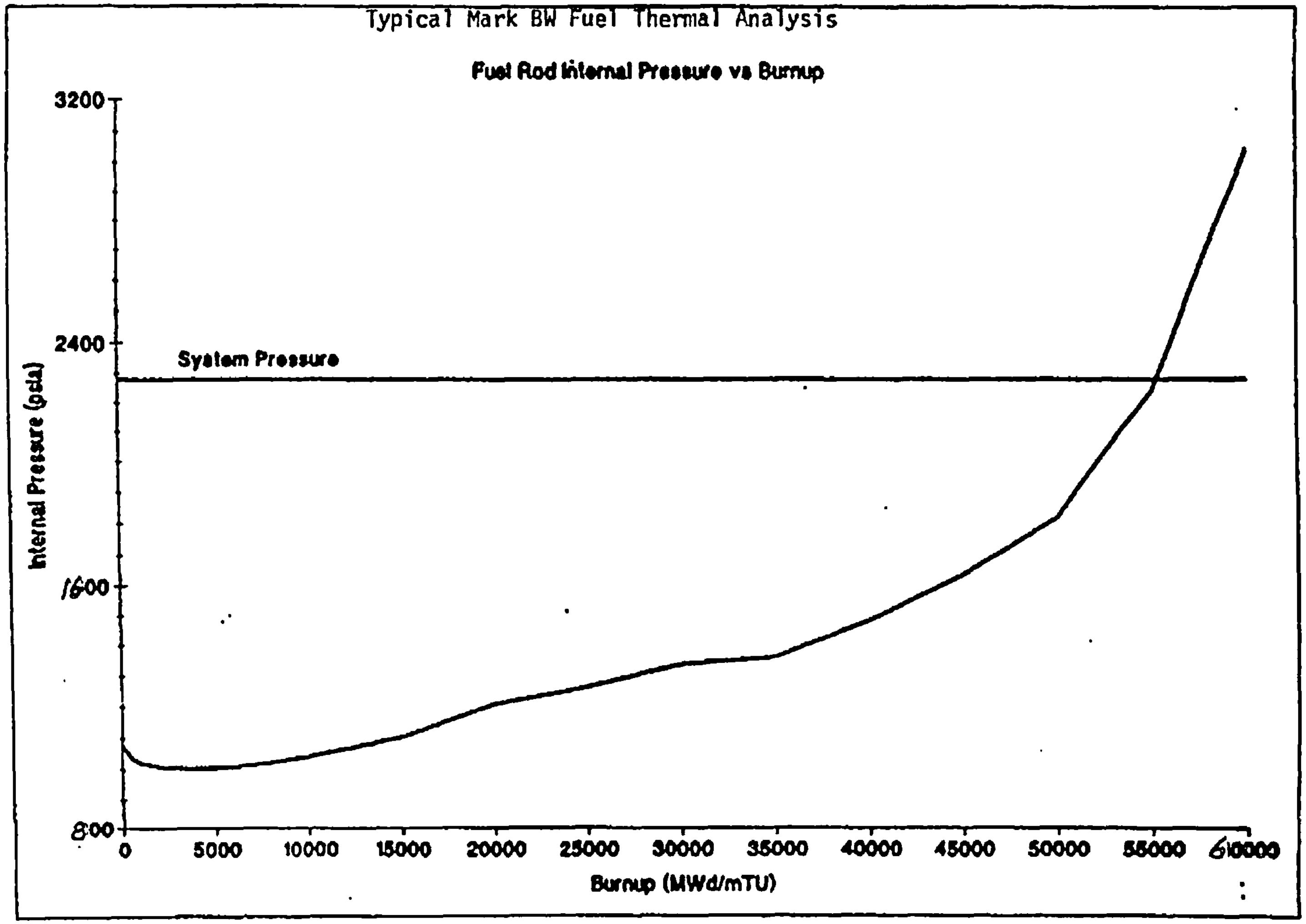

Typical Mark BW Fuel Thermal Analysis
Fuel Rod Internal Pressure vs Burnup
3200
2400
1600
800
System Pressure
Internal Pressure (psia)
0
5000
10000
15000
20000
25000
30000
35000
40000
45000
50000
55000
60000
Burnup (MWd/mTU)

FIGURE 5-5

6. MARK-BW NUCLEAR DESIGN EVALUATION

The Mark-BW is similar in design to the Westinghouse STD 17x17 assembly, except the Mark-BW contains spacer grids made of Zircaloy (instead of Inconel) that reduce parasitic neutron absorption. The Mark-BW assemblies have a larger diameter fuel rod which results in a higher loading of uranium and a lower hydrogen-to-uranium ratio (H/U) than the Westinghouse OFA design. On an equal enrichment basis the Mark-BW initially exhibits less reactivity than the OFA primarily because of less neutron thermalization. As exposure increases this difference diminishes and eventually the Mark-BW has the greater amount of reactivity due to the higher fuel loading and the harder neutron spectrum, which has bred greater amounts of fissile plutonium than the OFA. As a result the rate of depletion of reactivity is smaller for the Mark-BW than the OFA. This difference will not have any adverse impact on the operation of the plant.

The Doppler coefficients are not significantly different than the OFA and are essentially the same as the STD 17x17. Due to the lower H/U ratio the moderator coefficient for the Mark-BW is more negative than the OFA, but is not appreciably different than the moderator coefficient of the STD assembly. Previous fuel cycles of Westinghouse cores have contained mixtures of STD and OFA assemblies with acceptable power peaking and reactivity behavior.

From a physics standpoint, the Mark-BW assembly design is not a large change from previous Westinghouse assembly designs already licensed and operated. The use of the Mark-BW assembly design in the Westinghouse core either alone or in conjunction with the OFA and STD 17x17 does not adversely affect plant operation or neutronic parameters.

7. OVERALL IMPACT EVALUATION

The Mark-BW fuel assembly is compatible with both the Westinghouse STD and OFA designs. Special BWFC features incorporated into the design include the use of keyable grids which minimize scratching of the fuel rods during assembly, and a spacer grid restraint system which allows the grids to follow the fuel rods early in life. This will reduce axial friction during manufacture and during in-reactor operation.

The design of the Mark-BW fuel rod design is externally similar to the Westinghouse STD design. Design features different from both the STD and OFA designs include the used of thicker cladding, higher density fuel pellets, and the use of a double spring system to position the fuel stack. The manufacturing methods and texture of cladding for the Mark-BW is nearly identical to those of the cladding used for Mark-B production.

The design of the Mark-BW is based on technology proven by BWFC and NFI in operation in-reactor. Using methods similar to those used to analyze BWFC Mark-B design, the Mark-BW design has been analyzed and found to be acceptable.

Features of the Mark-BW design such as fuel rod length may change as the design develops. An analysis of any significant design features will be done and will be reported in the standard reload report along with other analyses that cannot be done until the fuel cycle design for each cycle is completed.

The Mark-BW and OFA designs are hydraulically similar with the differences in grid pressure drops and flow areas nearly offsetting each other. The net flow diversion between fuel assemblies in a transition core is small with acceptable effects on lift forces, crossflows, DNBR margins and transient response. Fuel rod thermal performance is acceptable up to 60,000 MWd/mtU provided that the linear heat rates remain below those shown in Figure 5-3.

The Mark-BW design will have a different rate of depletion of reactivity compared to the OFAs. The moderator coefficient is more negative compared to the OFA, but there are no significant differences in the Doppler coefficients. From a physics standpoint, the Mark-BW assembly design is not a large change from previous Westinghouse assembly designs already licensed and operated. The use of the Mark-BW assembly design in the Westinghouse core either alone or in conjunction with the OFA and STD 17x17 does not adversely affect plant operation or neutronic parameters.

The use of the Mark-BW fuel assembly is not expected to impact reactor plant margins or operation. Operational characteristics will vary slightly from the Westinghouse STD and OFA designs. These differences will be accounted for during transition cycles when the Mark-BW is resident with varying combinations of STD and OFA assemblies.

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Appendix A to BAW-10172

Responses (dated 6/2/89) to NRC Questions

QUESTIONS FOR MARK-BW MECHANICAL DESIGN REPORT (BAW-10172P)

Question 1: The subject document states in several instances that either particular analyses are still under evaluation or that data is to be obtained from four Mark-BW lead assemblies (LAs) being irradiated in McGuire Unit 1 in order to verify the Mark-BW design (such as analysis of LDV test results, assembly and rod growth data, oxidation data, qualification testing of the ferrite welding process, testing and analysis of the debris-resistant bottom nozzle, and post-irradiation cladding dimensional changes to verify the application of the B&W creep model to the Mark-BW design). Have any of these analyses been completed or have any data been obtained since the submittal of the subject document? Also, what is the irradiation and examination schedule for the Mark-BW LAs?

Response: When the Mark-BW Mechanical Design Topical was submitted for review a number of programs were underway to provide additional verification data for the design. The status of the relevant programs is given here.

a.) LDV Testing: The testing has been completed. Test data has been reduced and preliminary results are as expected. [ ] test configurations have been studied, involving bundle interior regions for both Zircaloy and Inconel spacer grids and bundle peripheral regions for grid interfaces. Preliminary results indicate that analytical predictions correlate well to experimental data. These results demonstrate that the Mark-BW grid yields acceptable flow characteristics when residing adjacent to fuel assemblies with either Inconel or Zircaloy mixing-vane grids.

In addition to the LDV tests, fuel assembly pressure drop tests have been performed on Westinghouse fuel assemblies of the type that will be resident during the introduction of the Mark-BW fuel. These tests were performed with BWFC's transportable flow test rig (TFTR). Measurements on a prototype

Mark-BW fuel assembly were first taken, followed by measurements on several Westinghouse production fuel assemblies. These measurements, which yielded pressure drop vs flow and flow-to-liftoff data, confirm the hydraulic compatibility of the Mark-BW with resident fuel.

b.) Assembly and rod growth data: The first cycle poolside post irradiation examination (PIE) of the Mark-BW lead asse blies (LAs) was completed in November 1988. Based on preliminary data evaluation the fuel assembly and fuel rod growth were both less than the design models. Fuel assembly and rod growth are shown on Figures 1 and 2.

c.) Oxidation data: Additional data that has been obtained since the submission of the Mark-BW design topical:

First cycle Mark-BW poolside PIE - 15 GWd/mtU assembly burnup
Third cycle Mark-BEB hotcell - 46 GWd/mtU assembly burnup

Preliminary data evaluation shows oxide thicknesses below the design values predicted for each burnup range. These are shown together with the Mark-B oxide data on Figure 3.

Data observations that are planned in the near future include:

Second and Third cycle Mark-BW poolside PIE
Fourth cycle Mark-GdB poolside PIE - 59 GWd/mtU

d.) Qualification testing of the restraining ferrule welding process: The spacer grid restraining scheme has been changed with an increase from 4 to 12 restraining guide tubes. Strength and corrosion qualification of the ferrule system on 12 guide tubes is in progress. Preliminary results show that the strength and corrosion resistance of the samples welds is adequate.

e.) Debris Resistant IEF: Testing has been completed on a debris resistant IEF for the Mark-BW. A pressure drop test was completed on a prototype fuel assembly with the debris resistant IEF. The difference in unrecoverable core pressure drop between a fuel assembly with the debris resistant IEF and one with the standard IEF is less than a [ ] increase. Structural analysis of the IEF was completed using a finite element code. The impact of using the debris resistant IEF on mechanical and thermal hydraulic performance is minimal.

f.) Post irradiation creep and oxidation data: Creepdown and oxidation data have been obtained from the Mark-BW IAs. Preliminary evaluations show that the data is in the general range expected as shown on Figures 3 and 4. The irradiation creep constant will be determined when a wide range of fluence is available. The oxide growth observed is within the envelope of previous data. However it has a high noise component (scatter) due to the small oxide thicknesses examined which are on the order of instrument error.

g.) Mark-BW IA PIE Schedule: Mark-BW IA post irradiation exams (PIEs) are expected to take place at a frequency of 14 to 16 months depending on plant capacity. The first cycle PIE took place in November 1989. A total of three PIEs are planned. The second is planned for January 1990, and the third for March 1991.

<u>Question 2:</u> It is presumed that the axial gaps between the top nozzle and the fuel rods, and between the top nozzle and reactor core plates for the Mark-BW design have been calculated based on data from Mark-B designs with annealed guide tubes. Is this presumption correct, and, if so, what is the burnup range of this data? Have these calculations of gap sizes been performed to a particular statistical confidence level, and how much margin is there between closure and calculated end-of-life gap size?

Response: Growth models for the Mark-BW fuel assembly and fuel rod are based on those used for Mark-B fuel designs. The models used provide a [ ] confidence that [ ] of the data are conservatively predicted. The fuel rod growth model is based on Mark-B data while the annealed guide tube growth model is based on industry and Mark-B data. The data for the models extends to [ ]. It will be reformulated with the availability of [ ] data from the Mark-GdB assembly which was recently discharged from Oconee 1. The fuel rod growth model was modified for use with the Mark-BW by moving the onset of accelerated growth down in burnup from [ ]. This was done to account for the faster creepdown of the Mark-BW fuel rod when compared to the Mark-B. The modified growth curve is shown in Figure 2.

Preliminary evaluation of first cycle PIE data for the Mark-BW IAs show that the growth is contained inside the model as shown on Figure 1. Based on nominal growth rates at [ ] the fuel assembly upper nozzle to reactor core plate gap will have the following margins to closure:

At 600 deg F, gap = [ ] inches
At 140 deg F, gap = [ ] inches

The growth rate tolerances and mechanical tolerances are combined by the square-root-sum-of-the-squares (SRSS) method. It is possible under worst case conditions that at shutdown temperatures of 100 deg F an interference of less than [ ] inches will occur. This degree of axial compression of a Mark-BW fuel assembly has been determined by testing to result in acceptable loads.

The Mark-BW fuel assembly design has an increased top nozzle to fuel rod gap when compared to the IAs of [ ] vs [ ] inches. This gives a nominal gap of [ ] inches at [ ]. Worst case gap under operating conditions (most conservative) is [ ] inches.

Question 3: In Section 4.1.1.2 it is stated that as a result of a particular operating condition, there will be a small amount of assembly liftoff. Is the possibility of liftoff unique to the Mark-BW assembly or is this possible for other B&W designs? Please discuss the degree of liftoff and its impact on core physics (reactivity and peaking factors) and thermal hydraulics.

Response: For condition I and II events no fuel assembly lift occurs. For the vertical LOCA and the [ ] pump overspeed transients some lift occurs, but the holddown spring is not compressed solid. A previous evaluation for a lifted fuel assembly done for the Mark-B design found no adverse effect on DNBR. The [ ] pump overspeed transient is not applicable to the Mark-B fuel assembly, and applies only to the Mark-BW design. The effect of potential liftoff on core physics and thermal hydraulics is insignificant.

Question 4: The inspections to be performed on the Mark-BW IAs are listed in Section 4.4.3 of the subject report and it is noted that fuel rod bow and fretting wear measurements on the fuel rods and guide tubes are omitted. Please provide justification for the omission of these measurements.

Response: Based on experience and the similarity between the Mark-BW and other BWFC fuel designs it was decided not to include measurements of fuel rod bow, fuel rod fretting and guide tube fretting in the IA PIE. Experience includes ex-reactor testing on the Mark-BW design and PIE exams on similar designs. The results of these tests and exams indicate that fuel rod bow and rod and guide tube fretting are not problem areas.

Fuel rod bow has been measured on the following designs, Mark-B with Inconel and Zircaloy intermediate spacer grids, Mark-C, and the Zirc LTAs for the Haddam Neck reactor. In all three designs, rod bow measurements supported the conclusions of the B&W Rod Bow Topical (reference 6), that no penalty is required on DNBR calculations due to rod bow, nor is rod to rod fretting wear expected.

Fuel rod fretting has been evaluated in ex-reactor loop tests of a Mark-BW prototype simulating in-reactor conditions, in Mark-B fuel rods with Inconel spacer grids though [ ], and in Mark-B fuel rods with Zircaloy intermediate spacer grids through [ ]. Although the contact points between the grid stops and the cladding were obvious, no depth was measurable. This indicates that fuel rod fretting is not a concern. The ex-reactor testing of the Mark-BW prototype included spacer grids with [ ] cells to simulate the maximum expected incore relaxation. Based on these results no impact on fuel rod integrity from fretting is expected.

Guide tube fretting was examined for B&W plants in reference 3. The guide tube fretting wear that was found was small and the structural integrity of the fuel assembly was not impacted. Mark-BW guide tube diameter dimensions are identical with the Westinghouse standard (17x17) assemblies, and the lengths are very slightly different. With dimensional similarity, the fretting response is assumed to be identical with the Westinghouse assemblies for which no problems were reported in reference 4.

Question 5: It is stated in Section 4.3.1 that an "advanced Zircaloy-4 cladding will be used on the Mark-BW fuel assemblies." Please submit information that documents the changes in this advanced cladding from that previously used by B&W. This information should include any alloying, thermal and chemical changes to the cladding, oxidation behavior, and mechanical property comparisons to the previous cladding material.

Response: Cladding development for the entire BWFC product line is an ongoing process with several demonstration assemblies irradiating a range of advanced cladding types. Planned full batch Mark-BW reload assemblies will use a cladding with the following changes from the current cladding used.

Decreased average tin from [ ] to [ ]
Increased intermediate annealing temperature by [ ] deg F.

These two changes will result in a lower oxidation rate, and similar creep properties compared to the current supply of cladding. Material properties are within the range of previous values, and will not require changes in properties used in computer codes.

Question 6: The discussion of cladding oxide growth in Section 4.5.1 states that a "maximum span value expected at high burnups" will be used for the maximum oxide layer thickness in Mark-BW analyses. Will the methodology and oxidation values presented in Appendix I of the TACO3 documentation be used for evaluating cladding oxidation in the Mark-BW design? If not, please define in greater detail how this maximum span value in Section 4.5.1 will be applied in the various licensing analyses, e.g., cladding stress and strain, LOCA, rod pressures, etc.

Response: The accounting for cladding oxide thickness depends upon the analysis and the version of TACO used for the analysis. Previous analyses and those currently underway use TACO2 and account for oxide thickness as shown in references 7 and 10, the TACO2 and extended burnup topicals. Analyses performed after the approval of TACO3 is obtained will follow Appendix I of the TACO3 topical (Reference 8). For the strain, fuel temperature inputs to LOCA, and rod pressure analyses, the cladding oxide thickness presented in Appendix I of the TACO3 topical will be used.

The maximum span value referred to in Section 4.5.1 is used in stress related calculation, i.e., stress intensity and fatigue usage factor calculations. In those calculations the minimum cladding thickness is further thinned by an amount equivalent to the thickness of base metal converted to oxide. The Mark-BW fuel rod will have adequate stress margins at a span maximum oxide thickness of [ ] inches.

Question 7: Please discuss the consequences of the projected reduction in cladding ductility at the high local burnups requested for the Mark-BW design in reference to the observed reductions in uniform cladding ductility observed in Reference 1 at a local burnup of [ ]

Response: The reduction in ductility reported in reference 1 and shown on Figure 5 is not considered representative. Hoop testing of cladding from the same fuel rods resulted in a larger uniform ductility (Figure 6, data from reference 2). Tensile testing of the Mark-BEB fuel rod cladding after 3 cycles and shown on Figure 7 (Ref 14) showed uniform ductility significantly greater than the [ ] limit imposed on fuel rod operation (Ref 5). Therefore any anticipated reduction in cladding ductility is not expected to adversely impact the Mark-BW fuel rod performance.

Question 8: It is stated in the cladding strain analyses (Section 4.5.3) that the fuel rod is ramped at a particular burnup level and with linear heat rate steps until the [ ] strain criteria is exceeded. Please explain how the particular burnup level and the linear heat rate steps are conservative in relation to B&W fuel operation and this analysis.

Response: The cladding strain analysis is currently performed with TACO2. The results are independent of fuel rod burnup. The analysis is performed following the method from reference 7 with the exception of the LHR. Because of the smaller pellet-cladding gap of the Mark-BW fuel rod compared to the Mark-B [ ], the [ ] strain limit will be reached prior to fuel melting. The strain is based on the pellet diametral expansion which results in a larger strain than the change in cladding outside diameter. Other conservatisms include the use of an oversized pellet diameter as input into TACO2. This pellet diameter is greater than that expected at [ ]. The step ramp approximates a single ramp as the burnup steps between ramps are very small. The purpose of the small steps is to ensure code stability and to accurately determine when [ ] strain is reached. The ramp is conservative as it begins at a LHR of [ ] The cladding strain limit of [ ] is exceeded at a peak LHR of [ ].

Strain analyses done after TACO3 approval is obtained will use the method presented in Appendix I of the TACO3 reference. The fuel rod will be ramped until a cladding strain of [ ] is reached. This analysis is not independent of burnup, and will be performed at various burnups to establish a kW/ft vs burnup limit for cladding strain.

<u>Question 9:</u> The acceptance criteria for creep collapse is somewhat vague in Section 4.5.5. What degree of cladding ovality does B&W consider to be the collapse point in the creep collapse lifetime calculation? Please provide further justification for the worst case densification values for [ ] TD fuel in Section 4.5.5. If there are any changes in the creep collapse methodology from that previous approved, please justify them.

<u>Response:</u> Ovality alone does not determine the collapse of the cladding. The ovality, cladding thickness, temperature, and differential pressure across the cladding together determine when the first of two possible conditions occur. When either of these two conditions occur the cladding is assumed to have failed. The conditions are:

1. The rate of ovalization equals or exceeds [ ].
2. The maximum stress level in the cladding equals or exceeds the [ ].

For the Mark-BW fuel rod condition 2 sets the predicted creep collapse life.

The maximum densification values assumed for [ ] TD fuel are based on production experience. In core densification can be predicted from resintering density changes after [ ]. Based on these the maximum densification expected is [ ] at [ ]. This results in a lower pin pressure than using a maximum density of [ ] TD at [ ] as is used with [ ] TD Fuel and is therefore conservative.

Except for the densification assumptions for [ ] TD fuel all other parts of the creep collapse analysis follows the methods demonstrated in references 7, 9 and 10.

<u>Question 10:</u> The power history envelope for the Mark-BW design in Figure 5-3 does not match the envelope for this design provided in Figure I-13 of Appendix I in the TACO3 documentation. Why is there a difference between these power history envelopes? How is the power history envelope in Figure 5-3 determined? Will the power history envelope in Figure 5-3 be used for internal rod pressure analyses? Will the rod pressures for the Mark-BW design exceed system pressure at the maximum burnups requested?

<u>Response:</u> The power history envelope for the Mark-BW design shown in Figure 5-3 of BAW-10172P was developed using methods described in BAW-10141P-A, rev. 1, and BAW-10153P-A. This power history was used in calculations performed with TACO2 to produce the fuel temperature and internal pressure predictions shown in Figures 5-4 and 5-5 of BAW-10172P. The power history envelope shown in Figure I-13 of the TACO3 documentation (8) represents a later update, developed with a slightly different method (i.e. uncertainty adjustments are treated differently). The application of this envelope to the analysis of fuel performance for any specific fuel cycle is subject to the same criterion as the TACO2 envelope, namely that it must be demonstrated to bound the predicted fuel rod power vs burnup for the most limiting rods in the core. All future Mark-BW fuel rod thermal performance analyses will use TACO3. This envelope, or similar envelopes developed with the same procedure, will be used for the prediction of fuel rod internal

pressure vs burnup. Until separately justified, Mark-BW fuel rod burnup will not be permitted to exceed the value for which the predicted internal pressure becomes greater than the nominal (core exit) reactor coolant system pressure.

Question 11: The NRC Standard Review Plan requirements for fuel enthalpy, fuel rod ballooning and rupture, cladding embrittlement and cladding melting have not been addressed in the subject document. Please supply the analyses that address these requirements or state where these analyses are to be addressed in future documentation.

Response: The SRP requirements for fuel enthalpy are related to the rod ejection accident for PWRs. The rod ejection accident is discussed in section 4.4.8 of reference 11. Fuel rod ballooning and rupture models are provided in section 4.3.3 of reference 12. Cladding embrittlement and melting will be addressed in the Catawba and McGuire LOCA applications topical report which is scheduled to be submitted to the NRC on August 31, 1989.

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FIGURE 1

FUEL ASSEMBLY GROWTH

ANNEALED GUIDE TUBES

FIGURE 2

A-15

FIGURE 3

Maximum Span-Average Oxide Thickness (Average and Range) Vs Burnup Hot Cell Data on Three-, Four-, and Five-Cycle Rods

Figure 4, Span Six Average Diameter
Span Six - Maximum Creepdown Span

Figure 5

Cladding Elongation at 650F Vs Burnup -- Axial Tension Tests, 5 Cycle Mark-B

Figure 6

Cladding Elongation at 650F Vs Burnup -- Ring Tension Tests, 5 Cycle Mark-B

Figure 7

TOTAL AND UNIFORM ELONGATION RESULTS

Appendix B to BAW-10172

Responses (dated 9/7/89) to NRC Questions

Question:

Provide bounding results of a mixed core seismic and LOCA analysis consisting of Westinghouse and Mark-BW Fuel assemblies.

Response:

A bounding analysis of a mixed core configuration (Mark BW and Westinghouse fuel assemblies) representative of a transition cycle was performed. The properties of the Westinghouse assemblies required for this bounding mixed core analyses were provided by Duke Power Company. Both Westinghouse and Mark-BW fuel assemblies have essentially the same overall assembly stiffness and frequency characteristics. Hence identical assembly stiffnesses were assumed in the analysis. A comparison of the Mark-BW and the Westinghouse grid properties showed that the stiffness of the Westinghouse grid is about [ ] higher than that of the Mark-BW grid. As the Westinghouse grid is stiffer than the Mark-BW grid, there will be a reduction in spacer grid impact loads on the Westinghouse spacer grid for a mixed core configuration compared to an all Westinghouse fueled core.

A reactor core pattern as shown in Figure 1 was analyzed for the reload transitions core. In this core pattern, [ ] are placed in the [ ] while the Westinghouse assemblies are placed in the [ ]. This pattern is conservative in the impact loading analysis, since [ ].

This configuration with the stiffer Westinghouse grid resulted in a maximum [ ] higher grid impact load on the Mark-BW fuel assembly compared to the full Mark-BW core configuration. This higher load resulted in a grid deformation of [ ] inches which is less than the limiting grid deformations. These are: [ ] inches deformation used for defining a coolable geometry and [ ] inches deformation used for assuring control rod insertability. The smaller of these two limits, [ ] inches for a coolable geometry, is used to determine margins. This results in a margin of [ ] [ ]. Hence it is concluded that a mixed core consisting of Westinghouse and BWFC Mark-BW fuel assemblies will meet the appropriate criteria and maintain a coolable geometry under combined seismic and LOCA accidents.

FIGURE 1 REACTOR CORE FUEL ASSEMBLY
RELOAD PATTERN CONSIDERED
IN THE ANALYSIS