April 7, 2004

MEMORANDUM TO: File

FROM: Michelle C. Honcharik, Project Manager, Section 1 **/RA/**
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: FRAMATOME ANP (FANP) – FACSIMILE TRANSMISSION OF DRAFT SAFETY EVALUATIONS AND E-MAIL REGARDING PROPRIETARY CONTENT (TAC NOS. MB7550 AND MB7551)

The attached draft safety evaluations (SE) for Topical Reports BAW-10239(P) and BAW-10238(P) were faxed to Ms. Gayle Elliot of FANP. She stated in the attached e-mail that neither draft SE contained proprietary information. The draft SEs can therefore be made publicly available.

Project No. 728

Attachments:
1. Draft SE for BAW-10239(P)
2. Draft SE for BAW-10238(P)
3. E-mail dtd. 3/24/04 from FANP

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DATE	4/6/04	4/6/04	4-7-04	*2/19/04 for BAW-10239 and 3/1/04 for BAW-10238

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BAW-10239(P)
"ADVANCED MARK-BW FUEL ASSEMBLY MECHANICAL DESIGN TOPICAL REPORT"

FRAMATOME ANP

PROJECT NO. 728

1.0 INTRODUCTION

By letters dated April 30, 2002, May 9, 2003, and **xxx, 2004**, (Refs. 1, 2 and 3), Framatome ANP requested approval of Topical Report BAW-10239(P), Revision 0,"Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report." Approval would permit licensees with Westinghouse three- and four-loop reactors that use a 17 x 17 fuel rod array to reference the generic topical report for use of the Framatome Advanced Mark-BW fuel. The Framatome ANP Advanced Mark-BW design is an evolution of the Mark-BW fuel design and includes new design features. This topical report evaluated the performance of the Advanced Mark-BW fuel design against the design criteria defined in the Standard Review Plan (SRP) Section 4.2, (Ref. 4).

In this topical report, Framatome requested approval for making minor changes to the fuel assembly design which would be evaluated against the criteria contained in this topical report and requested approval for making these design changes without specific NRC review and approval. Framatome also requested approval to have the plant specific evaluations of the performance of the Advanced Mark-BW fuel assembly not be required to be submitted to the NRC. The staff denies both of these requests at this time. The staff will continue to communicate with Framatome on mechanisms to approve minor changes to the fuel assembly design based on a methodology.

The staff approves use of this topical report subject to the following conditions:

1) This fuel assembly design is approved for use with Low Enriched Uranium Oxide (LEU) fuel.

2) The Advanced Mark-BW fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 MWD/MT.

3) Licensees referencing this topical must submit the plant specific evaluations of the fuel to the NRC for review and approval.

2.0 REGULATORY EVALUATION

Fuel designs must ensure that the reactor core will have appropriate margin to assure that the Specified Acceptable Fuel Design Limits (SAFDL) criteria in Title 10 Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 10 are met. Additionally, GDC 27 and 25 require that licensees maintain control rod insertability and core coolability. Loss-of-Coolant accident coolability requirements are contained in 10 CFR 50.46. The staff review process for new fuel designs is contained in Standard Review Plan (SRP) 4.2, Fuel System Design.

ATTACHMENT 1

The guidance provided within SRPs forms the basis of the staff's review and ensures that GDC 10 of 10 CFR Part 50, Appendix A is met.

3.0 TECHNICAL EVALUATION

The Framatome ANP Advanced Mark-BW fuel assembly design is an evolution of the Framatome ANP Mark-BW fuel assembly design and incorporates new design features including: the TRAPPER bottom nozzle, Mid-Span Mixing Grids (MSMGs), a floating intermediate grid design, a quick disconnect top nozzle, and use of M5 material for the cladding, structural tubing, and grids.

In this topical report, Framatome requested approval for making minor changes to the fuel assembly design which would be evaluated against the criteria contained in this topical report and requested approval for making these design changes without specific NRC review and approval. Framatome also requested approval to have the plant specific evaluations of the performance of the Advanced Mark-BW fuel assembly not be required to be submitted to the NRC. The staff denies both of these requests at this time. The staff will continue to communicate with Framatome on mechanisms to approve minor changes to the fuel assembly design based on a methodology.

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 fuel system is "not damaged "when fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analyses. Fuel rod failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has been breached. Coolability which is sometimes termed coolable geometry, means that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after an accident.

3.1 FUEL ASSEMBLY DESIGN

The Advanced Mark-BW fuel assembly design is intended for use in Westinghouse three- and four-loop reactors which use a 17 x 17 fuel rod array. The design is based on the Mark-BW fuel assembly design and incorporates some additional features that have been proved through reactor use such as: the TRAPPER bottom nozzle, Mid-Span Mixing Grids (MSMGs), a floating intermediate grid design, and, a low pressure drop quick disconnect top nozzle. The design also uses the M5 advanced alloy which has been previously approved (Ref 7) for cladding, structural tubing and grids. A thorough description and schematic diagrams of the fuel assembly, fuel rod, intermediate grids, low pressure drop quick connect/disconnect top nozzle, guide thimble and instrumentation tubing, mid-span mixing grids, debris filter bottom nozzle and the materials used for each component are provided in section 3.0 of the topical report. Based on the content of the topical report, the staff concludes that a satisfactory description of the fuel assembly has been provided for this review.

3.2 LTA PROGRAM

The LTA program for confirming the irradiation behavior of the Advanced Mark-BW fuel assembly design used four LTA in locations where the LTAs saw near-peak core power conditions. The LTA were irradiated for three cycles in the North Anna Unit 1 reactor. During their core residency, two cycles were in high duty locations and during the third cycle the LTAs were placed on the core periphery, a hostile hydraulic environment. Post-irradiation examinations (PIEs) were performed after every irradiation cycle to confirm that the LTAs were operating as predicted. The PIEs performed were appropriate for confirming the performance of the fuel design and the results met expectations; therefore, the LTA performance is acceptable.

3.3 DESIGN EVALUATION

The fuel system design bases must reflect these four objectives: 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. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. The design bases for each criteria remains the same as defined in the Mark-BW fuel assembly, ref 8.

3.3.1 FUEL SYSTEM DAMAGE CRITERIA

The design criteria relating to the fuel system damage should not be exceeded during normal operation including anticipated operational occurrences (AOOs). Fuel rod failure should be precluded and fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Each damage mechanism listed in SRP 4.2 will be reviewed to confirm that the design criteria is not exceeded during normal operation for the Advanced Mark-BW design.

3.3.1.1. STRESS

The design criteria for stress is that stress intensities for Advanced Mark-BW fuel assembly components shall be less than the stress limits based on the American Society of Mechanical Engineers (ASME) Code, Section III criteria, ref 9. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this stress criteria is acceptable for application to the Advanced Mark-BW fuel design.

The stress analyses for the Advanced Mark-BW fuel assembly design were reviewed as part of the staff audit of the topical report. The staff confirmed that for the stress analyses calculations, the worst case values in combination with the most limiting cycle conditions and most limiting transients were used to generate the most conservative results for each calculation. This deterministic method to obtain the most limiting stress value provides the most conservative stress value for each fuel assembly component. Positive margin to the design criteria was shown for each of the fuel assembly components; therefore the staff concludes that the fuel assembly design satisfies the design criteria for design stress.

3.3.1.2. CLADDING STRAIN

The design criteria for strain is that the Advanced Mark-BW fuel rod transient strain (elastic plus plastic) limit is 1% for Condition I and II events. This criteria is intended to preclude excessive cladding deformation during normal operation and AOOs. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this strain criteria is acceptable for application to the Advanced Mark-BW fuel design.

The analysis of the cladding strain uses the approved TACO3 code (ref. 10) to determine the cladding strain by evaluating the cladding circumferential changes before and after a linear heat rate (LHR) transient. The 1% strain limit corresponds to a transient LHR that is greater than the maximum transient the fuel rod is expected to experience for Condition I and II events. Therefore, the staff concludes that the fuel assembly design criteria for cladding strain are met.

3.3.1.3. CLADDING FATIGUE

The design criteria for cladding fatigue is that the Advanced Mark-BW maximum fuel rod fatigue usage factor is 0.9. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this cladding fatigue criteria is acceptable for application to the Advanced Mark-BW fuel design.

The methodology used for determining the cladding fatigue is outlined in ref 7. The methodology used a fuel rod life of 8 years and a vessel life of 20 years; therefore, the fuel rod will experience 20% of the number of transients that the vessel will. The analysis used all the Condition I and II events and one Condition III event to determine the total cladding fatigue usage factor. The maximum fatigue usage factor was determined to be well below the design criteria limit. Since the methodology is consistent with SRP 4.2 guidance and the maximum fatigue is well below the design criteria limit, the vendor has demonstrated that the cladding fatigue acceptance criteria has been met.

3.3.1.4. FRETTING

The design criteria for fretting is that the Advanced Mark-BW fuel assembly design shall be shown to provide sufficient support to limit fuel rod vibration and clad fretting wear and that span average cross-flow velocities shall be less than 2 ft/sec. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this fretting criteria is acceptable for application to the Advanced Mark-BW fuel design.

Generic mixed core analysis demonstrated that span average cross flows are less than the 2 ft/sec criterion. Therefore, the vendor has demonstrated that the Advanced Mark-BW fuel has the ability to meet this criterion.

The Advanced Mark-BW fuel assembly design is an evolution of the Mark-BW fuel assembly design. The in-reactor performance of the Mark-BW fuel assembly design shows that it has positive performance in the fretting arena. This performance is demonstrated through the use of the large number of fuel rods. Similarly, the in-reactor performance of the Advanced Mark-BW LTAs produced positive results even when the LTAs were subjected to the hostile hydraulic environment of the core periphery. The fuel vendor also performed out of core testing on the

Advanced Mark-BW fuel assembly design including a 1000 hour endurance test. The results of the endurance test demonstrated that the fuel rod wear was comparable to other currently approved fuel assembly designs. The staff concludes that the tests and data demonstrate that the Advanced Mark-BW fuel assembly design meets the design criteria for fretting.

3.3.1.5. OXIDATION, HYDRIDING, AND CRUD BUILDUP

The design criteria for oxidation, hydriding, and crud buildup is that the Advanced Mark-BW fuel rod cladding best-estimate corrosion shall not exceed 100 microns. This criteria is intended to preclude potential fuel system damage mechanisms. The SRP does not specify 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. Framatome accounts for the corrosion based on a database established for the M5 cladding material from in-reactor performance. This is acceptable because it uses realistic data that is representative of the material and burnup limits of the Advanced Mark-BW fuel assembly design.

Based on the data for M5 cladding material under prototypical irradiation conditions, the oxidation and hydrogen pickup rates are well below the criteria limit. Because crud is included as part of the oxidation measurement, the crud is also limited and well within the total acceptable range. Therefore, the vendor has demonstrated that the oxidation, hydriding, and crud buildup for the Advanced Mark-BW fuel assembly design has met the acceptance criteria.

3.3.1.6. FUEL ROD BOW

The design criteria for fuel rod bow is that fuel rod bowing shall be evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly. There is not a specific limit specified for fuel rod bow in SRP 4.2, the SRP only requires that rod bow be included in the design analysis.

The Framatome methodology for fuel rod bow was approved in ref 11. The database used to support this methodology was extended in ref 12. This database is representative of Zircaloy clad fuel. Because M5 cladding grows at a lower rate under irradiation conditions, the database for Zircaloy is conservative relative to the M5 performance. Therefore, the staff concludes that use of this database for predicting the rod bow of M5 clad fuel and continuing use of the penalty generated by the Zircaloy database for M5 fuel is conservative and acceptable for use.

3.3.1.7. AXIAL GROWTH

The design criteria for axial growth is that the Advanced Mark-BW fuel assembly-to-reactor internals gap allowance and the fuel assembly top nozzle-to-fuel rod gap allowance shall be designed to provide positive clearance during the assembly lifetime. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this axial growth criteria is acceptable for application to the Advanced Mark-BW fuel design.

The axial growth calculations were reviewed as part of the staff audit of the Advanced Mark-BW fuel design. The audit showed that the evaluation of the Advanced Mark-BW fuel design used approved M5 growth models and the worst case scenarios for calculating the clearances. The tolerances were combined in an appropriate manner and treated consistently. The lowest clearance values were obtained at EOL and in all evaluations, positive clearance remained at EOL under the worst conditions. Therefore, the staff concludes that the Advanced Mark-BW fuel design meets the axial growth acceptance criteria.

3.3.1.8. FUEL ROD INTERNAL PRESSURE

The design criteria for fuel rod internal pressure is that the fuel system will not be damaged due to excessive internal pressure. Fuel rod internal pressure is limited to that which would cause 1) the diametral gap to increase due to outward creep during steady-state operation and 2) extensive DNB propagation to occur. This design criteria was approved in ref. 13 and will continue to be valid since it is fuel design independent. therefore, this criteria is acceptable for application to the Advanced Mark-BW fuel design.

The fuel rod internal pressure analysis uses the TACO3 code with the methodology approved in ref. 13. This analysis performed on a plant specific basis includes the use of the most limiting manufacturing variations and a bounding power history for that plant. If the bounding analysis does not meet the fuel rod internal pressure criteria then on a cycle specific basis, a rod specific analysis using the actual power history and manufacturing data for that rod can be performed to demonstrate that the internal rod pressure criteria is satisfied. These dual analysis paths using the approved methodology are acceptable for use because they will demonstrate that the fuel rod internal pressure criterion is met.

3.3.1.9. ASSEMBLY LIFTOFF

The design criteria for assembly liftoff is that the Advanced Mark-BW fuel holddown springs must be capable of maintaining fuel assembly contact with the lower support plate during normal operating, Conditions I and II events, except for the pump overspeed transient. The fuel assembly shall not compress the holddown spring to solid height for any Conditions I and II event. The fuel assembly top and bottom nozzles shall maintain engagement with reactor internals for all Conditions I through IV events. This design criteria is consistent with the acceptance criteria of SRP 4.2 except for the exclusion of the pump overspeed transient. However, Framatome has been previously approved to exclude this transient; therefore, this assembly liftoff criteria is acceptable for application to the Advanced Mark-BW fuel design.

The vendor performed the analyses using the approved LYNXT code, ref 14, and, demonstrated that during all conditions considered except for the pump overspeed transient, the fuel assembly lift off criteria are met. During the pump overspeed transient, the lift is small and the holddown spring deflection is less than the worst-case normal operating cold-shutdown condition. The holddown spring is not compressed to a solid height for any operating condition. The audit verified that under all operating conditions, the top and bottom nozzles remained engaged with the reactor internals. Therefore, the staff concludes that for the Advanced Mark-BW fuel assembly design, the fuel assembly lift off criteria are met.

3.3.2 FUEL ROD FAILURE CRITERIA

The design criteria relating the fuel rod failure are applied in two ways. When they are applied to normal operation including AOOs, they are used as limits (SAFDLs) since fuel failure should not occur. When they are applied to postulated accidents, fuel failures are permitted and must be accounted for in the fission product releases. Fuel rod failure is defined as the loss of fuel rod hermeticity. Each fuel rod failure mechanism listed in SRP 4.2 will be reviewed to confirm that the design criteria is not exceeded during normal operation and is properly accounted for during postulated accidents for the Advanced Mark-BW design.

3.3.2.1. INTERNAL HYDRIDING

The design criteria for internal hydriding is that internal hydriding shall be precluded by appropriate manufacturing controls. For the Advanced Mark-BW assembly design, hydriding is prevented by keeping the level of moisture and hydrogenous impurities within the fuel to very low levels. Framatome maintains the fabrication level for total hydrogen in the fuel pellets to a level that is lower than the SRP 4.2 value of 2 ppm. This design criteria is consistent with the acceptance criteria of SRP 4.2 and is acceptable.

Framatome maintains the low hydrogen levels in the fuel rod through manufacturing controls. These controls have resulted in zero failures from internal hydriding for previous fuel designs. Because these controls will remain in place for the Advanced Mark-BW fuel assembly design and the limits are lower than the SRP 4.2 values, the design criteria will continue to be met with the Advanced Mark-BW fuel assembly design.

3.3.2.2. CLADDING COLLAPSE

The design criteria for cladding collapse is that predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life. The SRP states that if axial gaps in the fuel pellet column occur due to densification, the cladding has the potential to collapse into a gap. Because of the large local strains that accompany this process, collapsed cladding is assumed to fail. Because the Framatome design criteria is consistent with the acceptance criteria of SRP 4.2, it is acceptable for application to the Advanced Mark-BW fuel assembly design.

Framatome uses their approved creep collapse methodology, ref 15, to determine the potential for creep collapse of the Advanced Mark-BW fuel assembly design. This methodology uses conservative values to determine the creep collapse life of the fuel rod. Creep collapse is assumed when either the rate of creep ovalization exceeds 0.1 mils/hr or the maximum fiber stress exceeds the unirradiated yield strength of the cladding. Based on these definitions of creep collapse, the creep collapse lifetime was shown to be greater than 62 GWD/MT. Therefore, the Advanced Mark-BW fuel assembly design is adequately designed to prevent creep collapse for a service life up to 62 GWD/MT.

3.3.2.3. OVERHEATING OF CLADDING

The design criteria for cladding collapse is that for a 95% confidence level, DNB will not occur on a fuel rod during normal operation and anticipated operational occurrences (AOOs). The SRP states that it has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR) are satisfied. Because the Framatome design criteria is consistent with the acceptance criteria of SRP 4.2, it is acceptable for application to the Advanced Mark-BW fuel assembly design.

Framatome uses two approved CHF correlations in the analysis of DNB occurrence for the fuel, refs. 16 and 17. These are the BWU-N and BWU-Z CHF correlations. The BWU-N correlation is used for non-mixing vane grids while the BWU-Z correlation is used with enhanced mixing vane grids. These correlations address the types of mixing vane grids used in the Advanced Mark-BW fuel assembly design. Therefore, these correlations meet the design criteria and are acceptable for use.

3.3.2.4. OVERHEATING OF THE FUEL PELLETS

The design criteria for overheating of the fuel pellets is that for a 95% probability at a 95% confidence level, fuel pellet centerline melting shall not occur for normal operation and AOOs. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, it is acceptable for application to the Advanced Mark-BW fuel assembly design.

SRP 4.2 states that this analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. Framatome uses the TACO3 computer code to determine the local LHR throughout the fuel rod lifetime that could result in centerline temperature predictions exceeding the limit. The typical generic fuel centerline melt LHR is higher then any expected LHR at BOL which is the most limiting time of the cycle. Therefore, this analysis demonstrated that for the Advanced Mark-BW fuel assembly design the acceptance criteria are met.

3.3.2.5. PELLET/CLADDING INTERACTION

There is no generally applicable criteria for pellet/cladding interaction failure in SRP 4.2. The two criteria that should be applied per SRP 4.2 is that the uniform strain of the cladding should not exceed 1% and fuel melting should be avoided. Since both of these criteria were addressed previously in this safety evaluation, the criteria for pellet/cladding interaction is satisfied and acceptable for the Advanced Mark-BW design.

3.3.2.6. CLADDING RUPTURE

There is not a specific design limit associated with cladding rupture other than the requirements in 10 CFR 50.46, Appendix K. The cladding rupture correlation and supporting data were reviewed and approved for LOCA ECCS analyses in ref 7. Because this correlation was developed specifically for use in analyzing M5 cladding, the use of this correlation will provide the appropriate cladding rupture evaluations for the Advanced Mark-BW fuel assembly design under accident conditions.

3.3.3 FUEL COOLABILITY

For postulated accidents in which severe damage might occur, core coolability must be maintained as required by GDC 27 and 35. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit the removal of residual heat.

3.3.3.1. CLADDING EMBRITTLEMENT

To meet the requirements of 10 CFR 50.46, as it relates to LOCA, acceptance criteria of 2200 degrees F on peak cladding temperature and 17% on maximum cladding oxidation must be met. Framatome demonstrated through high temperature oxidation and quenching tests that the M5 cladding can meet these limits. The data and analysis to support this conclusion was reviewed and approved in ref 6. Further ref 7 concluded that the Baker-Just correlation is conservative for determining high temperature M5 oxidation for LOCA analysis and; therefore, is acceptable for LOCA ECCS analyses. Since the Baker-Just correlation is conservative and is required per 10 CFR 50.46 Appendix K, this criteria will be met without any modification needed to the approved LOCA ECCS codes.

3.3.3.2. VIOLENT EXPULSION OF FUEL

In severe reactivity insertion accidents (RIAs), such as a rod ejection event, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. To limit the effects of an RIA, regulatory guide 1.77, ref. 18, recommends that the radially averaged deposition at the hottest axial location be limited to 280 cal/g. Ref 7 reviewed and approved the use of M5 cladding and found that the use of M5 cladding has little impact on fuel expulsion and failure. Because the use of M5 will not impact the RIA analysis, this criteria will be met without any modification needed to the approved RIA analysis methodology.

3.3.3.3. FUEL ROD BALLOONING

To meet the requirements of 10 CFR 50.46, as it relates to evaluating ECCS performance during accidents, burst strain and flow blockage caused by ballooning of the cladding must be accounted for in the analysis of the core flow distribution. Framatome developed new ballooning and flow blockage models for M5 cladding which were reviewed and approved in ref 7.

Since these models were developed specifically for use in analyzing M5 cladding, the use of these models will provide the appropriate fuel rod ballooning for the Advanced Mark-BW fuel assembly design.

3.3.3.4. FUEL ASSEMBLY STRUCTURAL DAMAGE FROM EXTERNAL FORCES

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. During these events, fuel system coolability should be maintained and damage should not be so severe as to prevent control rod insertion when required. The design criteria for fuel assembly structural damage from external forces is divided into three categories:

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

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

LOCA or SSE+LOCA - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the ECCS and safety analysis.
These design criteria are consistent with SRP 4.2 guidance; therefore, they are acceptable for application to the Advanced Mark-BW fuel assembly design.

Framatome used the methodology in refs. 19 and 20 to perform generic evaluations of the structural damage from external forces. These analyses considered the horizontal and vertical impacts on the fuel assembly. These analyses included generic evaluations of the impact on the Advanced Mark-BW fuel assembly design when it is located in a mixed core. Various core loading patterns and locations in the core were utilized for the mixed core analysis impact. The results showed that the combined loads on the Advanced Mark-BW fuel assembly were small enough that coolable geometry is always maintained. The analyses results demonstrated that coolable geometry can be maintained under all the analyzed conditions; therefore, the vendor demonstrated that the acceptance criteria are met.

4.0 CONCLUSION

The NRC staff reviewed the acceptance criteria and generic and proposed analysis methodology presented by Framatome in the topical report "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report and determined that the criteria and proposed analysis methods are performed in accordance with the guidance provided in SRP 4.2. The staff finds the criteria and proposed analysis methods outlined in this topical report acceptable based on the determinations provided in the evaluation section of this safety analysis report and concludes that the topical is acceptable for referencing by licensees.

This evaluation supports the conclusion that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be adverse to the common defense and security or to the health and safety of the public. Therefore, on the basis of the above review and justification, the staff concludes that the Framatome Advanced Mark-BW fuel assembly design is acceptable for use is Westinghouse 3 and 4 loop design reactors which use a 17 x 17 fuel rod array with LEU fuel subject to the conditions includes in this safety evaluation report.

5.0 Conditions

The staff approves use of this topical report subject to the following conditions:

1) This fuel assembly design is approved for use with Low Enriched Uranium Oxide (LEU) fuel.

2) The Advanced Mark-BW fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 MWD/MT.

3) Licensees referencing this topical must submit the plant specific evaluations of the fuel to the NRC for review and approval.

6.0 REFERENCES

3. Letter from James Mallay, Framatome ANP to the USNRC, “Request for Review of BAW-10239(P), Revision 0, Adavnced Mark-BW Fuel Assembly Mechanical Design Topical Report,” April 30, 2002.

4. Letter from James Mallay, Framatome ANP to the USNRC, "Response to RAI to Topical Report BAW-10239(P)," May 9, 2003.

5. Letter from James Mallay, Framatome ANP to the USNRC, "this would be the letter for agreeing to the conditions," xxxxx, 2004.

6. NUREG-0800, Standard Review Plan 4.2, Fuel system Design, Revision 2, U.S. Nuclear Regulatory Commission, July 1981.

7. Letter from James Mallay, Framatome ANP to the USNRC, "Interim Report of an Evaluation of a Deviation Pursuant to 10 CFR 21.21(a)(2)," December 9, 2003.

8. EMF-92-116(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," February 1999.

9. BAW-10227-P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000.

10. BAW-10172P, "Mark-BW Mechanical Design Report," July 1988.

11. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1992.

12. BAW-10162-P-A, "TACO3 Fuel Pin Analysis Computer Code," October 1989.

13. BAW-10147-P-A, Revision 1, "Fuel Rod Bowing in Babcock & Wilcox Fuel Designs," May 1983.

14. BAW-10186-P-A, Revison 1, Supplement 1, "Mark-BW Extended Burnup," November 2001.

15. BAW-10183-P-A, "Fuel Rod Gas Pressure Criterion (FRGPC)," July 1995.

16. BAW-10156-P-A, Revision 1, "LYNXT: Core Transient Thermal-Hydraulic Program," August 1993.

17. BAW-10084-P-A, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," July 1995.

18. BAW-10199-P-A, "The BWU Critical Heat Flux Correlations," December 1994.

19. BAW-10199-P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," November 2000.

20. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

21. BAW-10133-P-A, "Mark-C Fuel Assembly LOCA-Seismic Analyses," June 1986.

22. BAW-10133-P-A, Revision 1, Addendum 1, "Mark-C Fuel Assembly LOCA-Seismic Analyses," October 2000.

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BAW-10238(P), REVISION 1 "MOX FUEL DESIGN REPORT"

FRAMATOME ANP

PROJECT 728

1.0 INTRODUCTION

By letters dated May 30, October 8, October 27, November 24,December 5, December 16, 2003 and March xx, 2004 (References 1 - 7), Framatome ANP requested approval of Topical Report BAW-10238(P), Revision 1, “MOX Fuel Design Report.” Approval of the topical report would permit the use of the fuel assembly mechanical design, Mark-BW/MOLX1, for MOX lead test assembly (LTA) use. The Framatome ANP Mark-BW/MOX1 fuel assembly design is an evolution of the Mark-BW fuel design and includes new design features. This topical report evaluated the performance of the Mark-BW/MOX1 fuel assembly design against the design criteria defined in the Standard Review Plan (SRP) Section 4.2. The technical evaluations contained in the subject topical report demonstrated that the Mark-BW/MOX1 fuel design met the criteria defined in SRP 4.2.

In addition, Framatome requested approval for making minor changes to the fuel assembly design that have been evaluated as described in this topical report and for making the subsequent design changes without specific NRC review and approval. Framatome also requested approval for batch implementation of the MOX fuel design. The staff denies these requests. Additionally, Framatome requested approval for using the LTAs in this topical report. This request is denied because approval for using LTAs is only granted to a specific licensee. Duke Energy Corporation (Duke) has requested approval for using the MOX LTAs in the Catawba plants. The staff will review the request for use of MOX LTAs in a reactor as part of the Duke request.

Framatome also requested approval for batch implementation of the MOX fuel design. This request is denied. Approval for batch loading of MOX fuel will only be granted to a specific licensee and must be requested in a separate licensing application by the licensee. The request for using batch loading of MOX fuel must be supported by the results from both the poolside and hot cell post irradiation examinations (PIEs).

Framatome included additional information in this topical report that was provided for information purposes and did not request approval for the areas the information covered. Therefore, anything contained in the topical report that is not explicitly approved in this SE has not been reviewed by the staff and is not acceptable. This includes but is not limited too the requested core loading fractions for MOX.

The staff approves use of this topical report subject to the following conditions:

1) This fuel assembly design is approved for use with Mixed Oxide fuel.

2) The Mark-BW/MOX1 fuel assembly design is licensed for LTA use only to a maximum fuel rod burnup of 60,000 MWD/MT.

3) The Mark-BW/MOX1 Fuel assembly design may not be modified from the design presented in BAW-10238.

4) The gallium content of the plutonium must be limited to 120 ppb.

5) The Uranium oxide powder used in the MOX fuel pellets must be fabricated by using the ADU process.

2.0 REGULATORY EVALUATION

Fuel designs must ensure that the reactor core will have appropriate margin to assure that the Specified Acceptable Fuel Design Limits (SAFDL) criteria in Title 10 Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 10 are met. Additionally, GDC 27 and 25 require that licensees maintain control rod insertability and core coolability. Loss-of-Coolant accident coolability requirements are contained in 10 CFR 50.46. The staff review process for new fuel designs is contained in Standard Review Plan (SRP) 4.2, Fuel System Design. The guidance provided within the SRPs forms the basis of the staff's review and ensures that GDC 10 of 10 CFR Part 50, Appendix A is met.

Title 10 of the Code of Federal Regulations (CFR) Part 50 Section 34, "Contents of Applications; Technical Information," requires that Safety Analysis Reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload process, licensees perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle.

There are no specific regulatory requirements or guidance available for the review of topical reports. As such, the staff review of this topical will be based on the evaluation of the experimental data provided to support the technical merit and compliance with any applicable regulations.

3.0 TECHNICAL EVALUATION

3.1 MOX DESIGN CONSIDERATIONS

During the development of the MOX fuel design, the vendor considered four areas where the properties of MOX fuel differ from current LEU fuel. These areas include: performance characteristics, plutonium content, MOX pellet homogeneity and microstructure, and operation in mixed cores.

The performance characteristics of MOX fuel are derived from the use of PuO_2 in a depleted uranium oxide matrix. Depleted uranium contains about 0.25% U235 with approximately 95% of the MOX pellet composed of UO_2.

The physical properties that may be affected by the addition of PuO_2 include:

- thermal conductivity
- thermal expansion
- thermal creep
- fission gas release
- in-reactor densification and swelling
- helium gas accumulation and release
- radial power profile
- melting point

These parameters have been investigated through experimental results and models to predict these parameters have been developed and incorporated into the COPERNIC computer code, reference 8. The staff approval of the COPERNIC code with the MOX parameters is contained in the staff safety evaluation related to reference 8 and will not be repeated here (see Reference 9).

In current LWR fuel, the fissionable component of the fuel is U235. For MOX fuel the fissionable components are Pu239 and Pu241. The isotopes Pu239 and Pu240 are present in low-enriched uranium fuel, although in a different concentration than in mixed oxide fuel. Therefore, some modifications of existing computer codes for the neutronic aspects of mixed oxide fuel is necessary to properly evaluate a reactor core containing both mixed oxide and low enriched uranium fuel. For the MOX LTAs, Duke Energy Corporation has described and evaluated the needed modifications to the neutronics code packages in Reference 10. The staff approval of these codes is contained in the related staff safety evaluation (Reference 11) and will not be repeated here.

Homogeneity of the MOX fuel is assured through the Micronized Master blend (MIMAS) manufacturing process. The MIMAS process produces fuel characterized by plutonium rich agglomerates dispersed in a uranium oxide matrix. The agglomerates consist of individual particles of PuO_2 and UO_2 powder. These agglomerates are controlled through the manufacturing process and verified through metallagraphic examination and/or autoradiography of fuel pellet samples from each batch of pellets. The fuel specification provides limits on the agglomerates. The US fuel specification is consistent with the current European specification therefore, the vendor has shown that the homogeneity of the MOX fuel for use in LTAs will not change from what is in use in Europe.

There are two types of mixed cores: thermal-hydraulic and neutronic. The thermal-hydraulic mixed core analysis uses Duke Energy Corporation approved methods, references 12-13. The neutronic mixed core analysis for the MOX LTAs uses the approved CASMO-4/SIMULATE-3 MOX codes, reference 10. The staff approval of these methods and codes is contained in the related staff safety evaluations and will not be repeated here.

3.2 WEAPONS-GRADE PLUTONIUM

The use of weapons-grade plutonium in MOX fuel is similar to the use of reactor-grade plutonium MOX fuel. However, some physical properties can be impacted by the differences between the two types of plutonium fuels. These differences include the isotopics, impurities, and pellet microstructure.

3.2.1 ISOTOPICS

The isotopic mixture between the weapons-grade and reactor-grade plutonium differs primarily because of how the different grades are produced. Reactor-grade plutonium is derived from spent LEU fuel that is reprocessed after being discharged from the core. Weapons-grade plutonium is irradiated for less time before being reprocessed. The difference in irradiation time affects the buildup of the plutonium 240, 241, and 242 isotopes. This difference is isotopes results in weapons-grade plutonium having a greater concentration of fissionable isotopes and lower concentration of absorber isotopes. This results in a decreased enrichment requirement for weapons-grade MOX fuel to achieve the equivalent burnup level and a different fuel reactivity change with burnup. This difference in isotopes and their depletion with burnup needs to be modeled explicitly by the neutronics code.

For the MOX LTAs, Duke Energy Corporation will use the approved CASMO-4/SIMULATE-3 MOX codes (Reference 10) which consider the effect of the weapons-grade MOX fuel isotopes to perform the core neutronic calculations. The staff approval of these codes is contained in the related staff safety evaluation (Reference 11) and will not be repeated here.

3.2.2 IMPURITIES

Weapons-grade plutonium contains gallium, a stabilizing element. However, quantities of gallium at the concentrations present in weapons-grade plutonium are undesirable for MOX fuel because gallium can cause material embrittlement. Therefore, weapons-grade plutonium will be polished to remove the gallium prior to the MOX processing stage. Polishing the plutonium will produce weapons-grade plutonium with a maximum gallium concentration of 120 ppb. This limit will be incorporated into the fuel specification.

Gallium is present in current LEU fuel in trace quantities. Archive samples of pellets and cladding material were evaluated by Oak Ridge National Laboratory (ORNL) to determine the level of gallium present in current LEU fuel. The evaluations determined that the current LEU fuel cladding contains 275 ppb of gallium. This level of gallium correlates to a pellet concentration of 50 ppb of gallium. The polishing process for the weapons-grade plutonium will limit the gallium to 120 ppb in the plutonium. This translates to a 10-20 ppb gallium level for the weapons-grade MOX fuel pellets. Therefore, the 120 ppb gallium limit will result in a final MOX fuel pellet that has a gallium concentration that is lower than current LEU fuel pellets. To date, a fuel failure from a gallium attack of the cladding has not been experienced. Therefore, using a gallium concentration limit that results in a final pellet concentration that is lower than current fuel is acceptable.

3.2.3 PELLET MICROSTRUCTURE

The use of weapons-grade plutonium instead of reactor-grade plutonium introduces slight differences into the fuel performance of the MOX fuel. These differences include the thermal conductivity, fission gas release, fuel pellet swelling, and pellet radial power distribution. These parameters used have been investigated and models to predict the parameters have been developed and incorporated into the COPERNIC computer code, reference 8.

3.3 MANUFACTURING PROCESSES

This section was provided for information purposes only as noted in the introduction section of the topical report. Therefore, nothing in this section is approved by this SER.

3.4 MARK-BW/MOX1 FUEL ASSEMBLY DESCRIPTION

The Mark-BW/MOX1 fuel assembly design is exactly the same as the Advanced Mark-BW fuel assembly design (Reference 14) with the exception that the Mark-BW/MOX1 design will use MOX fuel rods instead of Low Enriched Uranium (LEU) fuel rods. The Advanced Mark-BW fuel assembly design is approved for use in Westinghouse three- and four-loop reactors which use a 17 x 17 fuel rod array. The Mark-BW/MOX1 fuel assembly design incorporates the same features as the Advanced Mark-BW fuel assembly design including: the TRAPPER bottom nozzle, Mid-Span Mixing Grids (MSMGs), a floating intermediate grid design, a low pressure drop quick disconnect top nozzle, and use of the approved (Reference 15) M5 material for the cladding, structural tubing, and grids.

The Mark-BW/MOX1 fuel rod will be filled with MOX fuel pellets the entire stack length of the fuel rod. The fuel rod uses a stainless steel spring in the upper plenum to prevent the formation of axial gaps during shipping and handling. The MOX fuel pellets are designed in a manner consistent with uranium oxide pellets. They are chamfered at the top and bottom to facilitate pellets loading into the rods and are dish shaped at the ends. This geometery configuration will reduce the tendency of the pellet to change under irradiation into an hourglass shape.

There are four differences between the Advanced Mark-BW and Mark-BW/MOX1 fuel designs. To accommodate the additional fission gas release from the MOX fuel, the fuel rod is slightly longer due to an increase in the upper plenum volume. This change has an impact on the required shoulder gap which is discussed in the subject topical report. The fuel pellet density is decreased from 96% theoretical density to 95% theoretical density. This change was made so that the theoretical density would be consistent with the MOX pellet density currently is use in Europe. Similarly, the dish and chamfer design uses the European design instead of the American design. The fuel rod burnup will also differ and be lower than the approved burnup of uranium oxide fuel rods. The lower burnup is consistent with current European burnup limits.

A thorough description and schematic diagrams of the fuel rod, fuel assembly enrichment distribution, and comparison between the Advanced Mark-BW and the Mark-BW/MOX1 fuel assembly designs are provided in section 5.0 of the topical report. Based on the content of the topical report, the staff concludes that a satisfactory description of the fuel assembly has been provided for this review.

In this topical report, Framatome requested approval for making minor changes to the fuel assembly design which would be evaluated against the criteria contained in reference 14 and requested approval for making these design changes without specific NRC review and approval. The staff denies this request. This approval is for application of the Mark-BW/MOX1 fuel assembly design for use as LTAs; therefore, the LTA design must use the configuration and specifications submitted in this report without modification.

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 fuel system is ”not damaged“ when fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analyses. Fuel rod failure means that the fuel rod leaks and that the first fission product barrier (the cladding) has been breached. Coolability which is sometimes termed coolable geometry, means that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after an accident.

3.5 MARK-BW/MOX1 DESIGN EVALUATION

The fuel system design bases must reflect these four objectives: 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. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. The design bases for each criteria remains the same as defined in the Advanced Mark-BW fuel assembly, reference 14.

3.5.1 FUEL SYSTEM DAMAGE CRITERIA

The design criteria relating to the fuel system damage should not be exceeded during normal operation including anticipated operational occurrences (AOOs). Fuel rod failure should be precluded and fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. Each damage mechanism listed in SRP 4.2 will be reviewed to confirm that the design criteria is not exceeded during normal operation for the Mark-BW/MOX1 design.

3.5.1.1. STRESS

The design criteria for stress is that stress intensities for the Mark-BW/MOX1 fuel assembly components shall be less than the stress limits based on the American Society of Mechanical Engineers (ASME) Code, Section III criteria, reference 16. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this stress criteria is acceptable for application to the Mark-BW/MOX1 fuel design.

The stress analyses for the Advanced Mark-BW fuel assembly design were reviewed as part of the staff audit of the topical report. The staff confirmed that for the stress analyses calculations, the worst case values in combination with the most limiting cycle conditions and most limiting transients were used to generate the most conservative results for each calculation. This deterministic method to obtain the most limiting stress value provides the most conservative stress value for each fuel assembly component. The Mark-BW/MOX1 specific values were submitted in the subject topical report. Positive margin to the design criteria was shown for each of the fuel assembly components; therefore the staff concludes that the fuel assembly design satisfies the design criteria for design stress.

3.5.1.2. CLADDING STRAIN

The design criteria for strain is that the Mark-BW/MOX1 fuel rod transient strain (elastic plus plastic) limit is 1% for Conditions I and II events. This criteria is intended to preclude excessive cladding deformation during normal operation and AOOs. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this strain criteria is acceptable for application to the Mark-BW/MOX1 fuel design.

The analysis of the cladding strain uses the approved COPERNIC code (reference 8) to determine the cladding strain by evaluating the cladding circumferential changes before and after a linear heat rate (LHR) transient. The 1% strain limit corresponds to a transient LHR that is greater than the maximum transient the fuel rod is expected to experience for Condition I and II events. Therefore, the staff concludes that the fuel assembly design criteria for cladding strain are met.

3.5.1.3. CLADDING FATIGUE

The design criteria for cladding fatigue is that the Mark-BW/MOX1 maximum fuel rod fatigue usage factor is 0.9. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this cladding fatigue criteria is acceptable for application to the Mark-BW/MOX1 fuel design.

The methodology used for determining the cladding fatigue is outlined in reference 15. The methodology used a fuel rod life of 8 years and a vessel life of 20 years; therefore, the fuel rod will experience 20% of the number of transients that the vessel will. The analysis used all the Condition I and II events and one Condition III event to determine the total cladding fatigue usage factor. The maximum fatigue usage factor was determined to be well below the design criteria limit. Since the methodology is consistent with SRP 4.2 guidance and the maximum fatigue is well below the design criteria limit, it has been demonstrated that the cladding fatigue acceptance criteria has been met.

3.5.1.4. FRETTING

The design criteria for fretting is that the Mark-BW/MOX1 fuel assembly design shall be shown to provide sufficient support to limit fuel rod vibration and clad fretting wear and that span average cross-flow velocities shall be less than 2 ft/sec. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this fretting criteria is acceptable for application to the Mark-BW/MOX1 fuel design.

Mixed core analysis demonstrated that span average cross flows are less than the 2 ft/sec criterion. Therefore, it has been demonstrated that the Mark-BW/MOX1 fuel assembly meets this criterion.

Through extensive in-reactor use, the Mark-BW fuel assembly design has shown that it has positive performance in the fretting arena. The Mark-BW/MOX1 and Advance Mark-BW fuel assembly designs are an evolution of the Mark-BW fuel assembly design. The in-reactor performance of the Advanced Mark-BW LTAs produced positive results even when the LTAs were subjected to the hostile hydraulic environment of the core periphery. The fuel vendor also performed out of core testing on the Advanced Mark-BW fuel assembly design including a 1000 hour endurance test. The results of the endurance test demonstrated that the fuel rod wear was comparable to other currently approved fuel assembly designs. The staff concludes that the tests and data demonstrate that the Mark-BW/MOX1 fuel assembly design meets the design criteria for fretting.

3.5.1.5. OXIDATION, HYDRIDING, AND CRUD BUILDUP

The design criteria for oxidation, hydriding, and crud buildup is that the Mark-BW/MOX1 fuel rod cladding best-estimate corrosion shall not exceed 100 microns. This criteria is intended to preclude potential fuel system damage mechanisms. The SRP does not specify 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. Framatome accounts for the corrosion based on a database established for the M5 cladding material from in-reactor performance. This is acceptable because it uses realistic data that is representative of the material and burnup limits of the Mark-BW/MOX1 fuel assembly design.

Based on the data for M5 cladding material under prototypical irradiation conditions, the oxidation and hydrogen pickup rates are well below the criteria limit. Because crud is included as part of the oxidation measurement, the crud is also limited and well within the total acceptable range. Therefore, the vendor has demonstrated that the oxidation, hydriding, and crud buildup for the Mark-BW/MOX1 fuel assembly design has met the acceptance criteria.

3.5.1.6. FUEL ROD BOW

The design criteria for fuel rod bow is that fuel rod bowing shall be evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly. There is not a specific limit specified for fuel rod bow in SRP 4.2, the SRP only requires that rod bow be included in the design analysis.

The Framatome methodology for fuel rod bow was approved in reference 17. The database used to support this methodology was extended in reference 18. This database is representative of Zircaloy clad fuel. Because M5 cladding grows at a lower rate under irradiation conditions, the database for Zircaloy is conservative relative to the M5 performance. Therefore, the staff concludes that use of this database for predicting the rod bow of M5 clad fuel and continuing use of the penalty generated by the Zircaloy database for M5 fuel is conservative and acceptable for use.

3.5.1.7. AXIAL GROWTH

The design criteria for axial growth is that the Mark-BW/MOX1 fuel assembly-to-reactor internals gap allowance and the fuel assembly top nozzle-to-fuel rod gap allowance shall be designed to provide positive clearance during the assembly lifetime. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, this axial growth criteria is acceptable for application to the Mark-BW/MOX1 fuel design.

The axial growth calculations were reviewed as part of the staff audit of the Advanced Mark-BW fuel design. The audit showed that the evaluation of the Advanced Mark-BW fuel design used approved M5 growth models and the worst case scenarios for calculating the clearances. The tolerances were combined in an appropriate manner and treated consistently. The lowest clearance values were obtained at EOL and in all evaluations, positive clearance remained at EOL under the worst conditions. Therefore, the staff concludes that the Mark-BW/MOX1 fuel design meets the axial growth acceptance criteria.

3.5.1.8. FUEL ROD INTERNAL PRESSURE

The design criteria for fuel rod internal pressure is that the fuel system will not be damaged due to excessive internal pressure. Fuel rod internal pressure is limited to that which would cause 1) the diametral gap to increase due to outward creep during steady-state operation and 2) extensive DNB propagation to occur. This design criteria was approved in reference 19 and will continue to be valid since it is fuel design independent. therefore, this criteria is acceptable for application to the Mark-BW/MOX1 fuel design.

The fuel rod internal pressure analysis uses the COPERNIC code with the methodology approved in reference 8. This analysis includes the use of the most limiting manufacturing variations and a bounding power history for the cycles. The analysis demonstrated that margin remains throughout the fuel rod lifetime. Therefore, the staff concludes that the Mark-BW/MOX1 fuel assembly design meets the fuel rod internal pressure criterion.

3.5.1.9. ASSEMBLY LIFTOFF

The design criteria for assembly liftoff is that the Mark-BW/MOX1 fuel hold down springs must be capable of maintaining fuel assembly contact with the lower support plate during normal operating, Conditions I and II events, except for the pump overspeed transient. The fuel assembly shall not compress the hold down spring to solid height for any Conditions I and II event. The fuel assembly top and bottom nozzles shall maintain engagement with reactor internals for all Conditions I through IV events. This design criteria is consistent with the acceptance criteria of SRP 4.2 except for the exclusion of the pump overspeed transient. However, Framatome has been previously approved to exclude this transient; therefore, this assembly liftoff criteria is acceptable for application to the Mark-BW/MOX1 fuel design.

The vendor performed the analyses using the approved LYNXT code, reference 13, and, demonstrated that during all conditions considered except for the pump overspeed transient, the fuel assembly lift off criteria are met. During the pump overspeed transient, the lift is small and the hold down spring deflection is less than the worst-case normal operating cold-shutdown condition. The hold down spring is not compressed to a solid height for any operating condition. The audit verified that under all operating conditions, the top and bottom nozzles remained engaged with the reactor internals. Therefore, the staff concludes that for the Mark-BW/MOX1 fuel assembly design, the fuel assembly lift off criteria are met.

3.5.2 FUEL ROD FAILURE CRITERIA

The design criteria relating the fuel rod failure are applied in two ways. When they are applied to normal operation including AOOs, they are used as limits (SAFDLs) since fuel failure should not occur. When they are applied to postulated accidents, fuel failures are permitted and must be accounted for in the fission product releases. Fuel rod failure is defined as the loss of fuel rod hermeticity. Each fuel rod failure mechanism listed in SRP 4.2 will be reviewed to confirm that the design criteria is not exceeded during normal operation and is properly accounted for during postulated accidents for the Advanced Mark-BW design.

3.5.2.1. INTERNAL HYDRIDING

The design criteria for internal hydriding is that internal hydriding shall be precluded by appropriate manufacturing controls. For the Mark-BW/MOX1 assembly design, hydriding is prevented by keeping the level of moisture and hydrogenous impurities within the fuel to very low levels. Framatome maintains the fabrication level for total hydrogen in the fuel pellets to a level that is lower than the SRP 4.2 value of 2 ppm. This design criteria is consistent with the acceptance criteria of SRP 4.2 and is acceptable.

Framatome maintains the low hydrogen levels in the fuel rod through manufacturing controls. These controls have resulted in zero failures from internal hydriding for previous fuel designs. Because these controls will remain in place for the Mark-BW/MOX1 fuel assembly design and the limits are lower than the SRP 4.2 values, the design criteria will continue to be met with the Mark-BW/MOX1 fuel assembly design.

3.5.2.2. CLADDING COLLAPSE

The design criteria for cladding collapse is that predicted creep collapse life of the fuel rod must exceed the maximum expected in-core life. The SRP states that if axial gaps in the fuel pellet column occur due to densification, the cladding has the potential to collapse into a gap. Because of the large local strains that accompany this process, collapsed cladding is assumed to fail. Because the Framatome design criteria is consistent with the acceptance criteria of SRP 4.2, it is acceptable for application to the Mark-BW/MOX1 fuel assembly design.

Framatome uses their approved creep collapse methodology, reference 20, to determine the potential for creep collapse of the Mark-BW/MOX1 fuel assembly design. This methodology uses conservative values to determine the creep collapse life of the fuel rod. Creep collapse is assumed when either the rate of creep ovalization exceeds 0.1 mils/hr or the maximum fiber stress exceeds the unirradiated yield strength of the cladding. Based on these definitions of creep collapse, the creep collapse lifetime was shown to be greater than 60 GWD/MT. Therefore, the Mark-BW/MOX1 fuel assembly design is adequately designed to prevent creep collapse for a service life up to 60 GWD/MT.

3.5.2.3. OVERHEATING OF CLADDING

The design criteria for cladding collapse is that for a 95% confidence level, DNB will not occur on a fuel rod during normal operation and anticipated operational occurrences (AOOs). The SRP states that it has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR) are satisfied. Because the Framatome design criteria is consistent with the acceptance criteria of SRP 4.2, it is acceptable for application to the Mark-BW/MOX1 fuel assembly design.

Framatome uses two approved CHF correlations in the analysis of DNB occurrence for the fuel, refs. 21 and 22. These are the BWU-N and BWU-Z CHF correlations. The BWU-N correlation is used for non-mixing vane grids while the BWU-Z correlation is used with enhanced mixing vane grids. These correlations address the types of mixing vane grids used in the Mark-BW/MOX1 fuel assembly design. Therefore, these correlations meet the design criteria and are acceptable for use.

3.5.2.4. OVERHEATING OF THE FUEL PELLETS

The design criteria for overheating of the fuel pellets is that for a 95% probability at a 95% confidence level, fuel pellet centerline melting shall not occur for normal operation and AOOs. This design criteria is consistent with the acceptance criteria of SRP 4.2; therefore, it is acceptable for application to the Mark-BW/MOX1 fuel assembly design.

SRP 4.2 states that this analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. Framatome uses the COPERNIC computer code to determine the local LHR throughout the fuel rod lifetime that could result in centerline temperature predictions exceeding the limit. The fuel centerline melt LHR is higher then any expected LHR at BOL which is the most limiting time of the cycle. Therefore, this analysis demonstrated that for the Mark-BW/MOX1 fuel assembly design the acceptance criteria are met.

3.5.2.5. PELLET/CLADDING INTERACTION

There is no generally applicable criteria for pellet/cladding interaction failure in SRP 4.2. The two criteria that should be applied per SRP 4.2 is that the uniform strain of the cladding should not exceed 1% and fuel melting should be avoided. Since both of these criteria were addressed previously in this safety evaluation, the criteria for pellet/cladding interaction is satisfied and acceptable for the Mark-BW/MOX1 design.

3.5.2.6. CLADDING RUPTURE

There is not a specific design limit associated with cladding rupture other than the requirements in 10 CFR 50.46, Appendix K. The cladding rupture correlation and supporting data were reviewed and approved for LOCA ECCS analyses in reference 15. Because this correlation was developed specifically for use in analyzing M5 cladding, the use of this correlation will provide the appropriate cladding rupture evaluations for the Mark-BW/MOX1 fuel assembly design under accident conditions.

3.5.3 FUEL COOLABILITY

For postulated accidents in which severe damage might occur, core coolability must be maintained as required by GDC 27 and 35. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit the removal of residual heat.

3.5.3.1. CLADDING EMBRITTLEMENT

To meet the requirements of 10 CFR 50.46, as it relates to LOCA, acceptance criteria of 2200 degrees F on peak cladding temperature and 17% on maximum cladding oxidation must be met. Framatome demonstrated through high temperature oxidation and quenching tests that the M5 cladding can meet these limits. The data and analysis to support this conclusion was reviewed and approved in reference 15. Further reference 15 concluded that the Baker-Just correlation is conservative for determining high temperature M5 oxidation for LOCA analysis and; therefore, is acceptable for LOCA ECCS analyses. Since the Baker-Just correlation is conservative and is required per 10 CFR 50.46 Appendix K, this criteria will be met without any modification needed to the approved LOCA ECCS codes. This is shown in the LTA licensing application, reference 23.

3.5.3.2. VIOLENT EXPULSION OF FUEL

In severe reactivity insertion accidents (RIAs), such as a rod ejection event, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. To limit the effects of an RIA, regulatory guide 1.77, ref. 24, recommends that the radially averaged deposition at the hottest axial location be limited to 280 cal/g. Ref 15 reviewed and approved the use of M5 cladding and found that the use of M5 cladding has little impact on fuel expulsion and failure. Because the use of M5 will not impact the RIA analysis, this criteria will be met without any modification needed to the approved RIA analysis methodology. This is shown in the LTA licensing application, reference 23.

3.5.3.3. FUEL ROD BALLOONING

To meet the requirements of 10 CFR 50.46, as it relates to evaluating ECCS performance during accidents, burst strain and flow blockage caused by ballooning of the cladding must be accounted for in the analysis of the core flow distribution. Framatome developed new ballooning and flow blockage models for M5 cladding which were reviewed and approved in ref 15. Since these models were developed specifically for use in analyzing M5 cladding, the use of these models will provide the appropriate fuel rod ballooning for the Mark-BW/MOX1 fuel assembly design. The ability to meet this criteria is demonstrated in the LTA licensing application, reference 23.

3.5.3.4. FUEL ASSEMBLY STRUCTURAL DAMAGE FROM EXTERNAL FORCES

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. During these events, fuel system coolability should be maintained and damage should not be so severe as to prevent control rod insertion when required. The design criteria for fuel assembly structural damage from external forces is divided into three categories:

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

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

LOCA or SSE+LOCA - Ensure safe shutdown of the reactor by maintaining the overall structural integrity of the fuel assemblies and a coolable geometry within deformation limits consistent with the ECCS and safety analysis.

These design criteria are consistent with SRP 4.2 guidance; therefore, they are acceptable for application to the Advanced Mark-BW fuel assembly design.

Framatome used the methodology in references 25 and 26 to perform evaluations of the structural damage from external forces. These analyses considered the horizontal and vertical impacts on the fuel assembly. These analyses included evaluations of the impact on the Mark-BW/MOX1 fuel assembly design when it is located in a mixed core. Various core loading patterns and locations in the core were utilized for the mixed core analysis impact. The results showed that the combined loads on the Mark-BW/MOX1 fuel assembly were small enough that coolable geometry is always maintained. The analyses results demonstrated that coolable geometry can be maintained under all the analyzed conditions; therefore, the vendor demonstrated that the acceptance criteria are met.

3.6 EXPERIENCE BASE

MOX fuel was previously used in the U.S. prior to the 1977 decision to defer the commercial reprocessing and recycling of plutonium. An early experimental program at the Saxton reactor used plutonium with an isotopic content similar to current weapons grade plutonium. Seven assemblies were irradiated to a burnup of 51,000 MWD/MT and sent to hot cells for post-irradiation testing. The post-irradiation testing demonstrated that all rod failures were the result of cladding problems that were not dependent on the type of fuel material.

The LTA program for confirming the irradiation behavior of the Advanced Mark-BW fuel assembly design used four LTA with uranium oxide fuel in locations where the LTAs saw near-peak core power conditions. The LTA were irradiated for three cycles in the North Anna Unit 1 reactor. During their core residency, two cycles were in high duty locations and during the third cycle the LTAs were placed on the core periphery, a hostile hydraulic environment. Post-irradiation examinations (PIEs) were performed after every irradiation cycle to confirm that the LTAs were operating as predicted. The PIEs performed were appropriate for confirming the performance of the fuel design and the results met expectations; therefore, the LTA performance is acceptable. Because the Mark-BW/MOX1 fuel assembly design is the same as the Advanced Mark-BW fuel assembly design, the irradiation behavior of the LTA fuel assemblies is relevant to the expected Mark-BW/MOX1irradiation behavior.

MOX fuel has been used in Europe since the 1960's with the first irradiations performed in Belgium. MOX fuel using the current MIMAS fabrication process has been in use since 1987. The MIMAS process has been used to produce fuel pellets for 435,000 fuel rods. These fuel rods have reached burnup levels that are comparable to the requested 50 GWD/MT burnup limit. Additionally, European test assemblies have been irradiated to 60 GWD/MT and higher under core conditions which are similar to the proposed U.S. use without experiencing any fuel rod failures. Post-irradiation hot cell testing of MOX fuel rods demonstrated that the MOX fuel parameters were closely predicted by the analytical methods and that the fuel performance was similar to current LEU fuel performance. These irradiations and hot cell results demonstrate that MOX fuel can be used in LWRs up to the burnup limits requested without failure caused by the pellet material.

3.7 LEAD ASSEMBLY PROGRAM

Duke has requested approval for inserting four Mark-BW/MOX1 fuel assemblies into the Catawba reactor core in a separate application, reference 23. Mark-BW/MOX1 LTA use has multiple purposes including:

Demonstrating that the data and analytical models derived from the reactor-grade MOX fuel data are consistent with weapons-grade MOX behavior.

Demonstrating the applicability of the MIMAS process and impurity polishing process to the use of weapons-grade MOX fuel

Confirming that the lower trace levels of gallium do not impact the fuel cladding
Demonstrating the acceptable performance of the Mark-BW/MOX1 fuel assembly design

Demonstrating that the MOX fuel performance under U.S. PWRs operating conditions is acceptable

Confirming the validity of the neutronic models

3.7.1 FUEL EXAMINATIONS

To collect appropriate data to confirm the current models on the fuel performance and fuel behavior characteristics, Framatome developed a LTA inspection and testing program. This program consists of poolside post irradiation examinations (PIEs) and hot cell PIEs. The poolside PIEs will be performed between every cycle and following discharge from the core. Hot cell PIEs will be performed after 2 cycles of irradiation and after the third irradiation cycle.

Prior to irradiation, the fuel assemblies will be fully characterized. This will allow comparison between the PIE results and the original design. MOX fuel samples from each of the three plutonium loadings will be kept so that in the event of unexpected fuel performance, baseline material exists for root cause analysis.

Poolside PIEs will be performed between the irradiation cycles. These basic examinations will include fuel assembly visual, fuel rod visual, fuel assembly growth, fuel rod growth, and fuel assembly bow and distortion examinations. In addition, following core discharge a more extensive poolside PIE will be performed which includes grid width, fuel rod oxide thickness, grid oxide thickness, fuel assembly drag force, guide thimble plug gauge, and fuel rod bowing examinations.

Hot cell examinations will be performed after the second and third irradiation cycles. For these tests, fuel rods will be extracted from the fuel assemblies and shipped to a DOE laboratory for examination. These hot cell PIEs will include rod puncture, metallography/ceramography, cladding mechanical tests, burnup analysis, and burnup distribution examinations. These tests will validate analytical models that were developed for weapons-grade MOX fuel.

The staff agrees that these examinations will be needed to assess the fuel performance and fuel behavior of the MOX fuel under U.S. PWR reactor conditions. Therefore, the staff finds the planned minimum test matrix acceptable. The results of these PIEs will be necessary to support a future application for batch MOX fuel use.

3.8 APPLICABILITY OF PREVIOUSLY APPROVED METHODS TO MOX FUEL

The American Society of Mechanical Engineers (ASME) Code is used by the NRC for acceptance of mechanical analysis of nuclear power plant components (Reference 16). The code includes analysis methods for calculating structural integrity of materials. These methods depend on parameters such as temperature, type of material, and neutron fluence but they do not depend on the type of fuel material. Therefore, application of the ASME code to the analysis of MOX fuel is acceptable.

Topical Report BAW-10227P-A, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,” approved the use of the M5 material for LEU fuel (reference 15). This topical report included analysis methods that would be used for specific material parameters during fuel design evaluations. These material parameter analyses include cladding stress, cladding buckling, cladding fatigue, and axial growth. These parameters are all dependent on the material used and are not impacted by the fuel pellet material; therefore, use of the analysis methods in BAW-10227P-A is acceptable for use with MOX fuel.

Topical Report BAW-10156-A, “LYNXT: Core Transient Thermal-Hydraulic Program,” approved the LYNXT code for thermal-hydraulic core analysis (Reference 13). The LYNXT code is used in the analysis of fuel assembly liftoff and fuel rod fretting by calculating the cross flows developed between adjacent fuel assemblies. The thermal-hydraulics of the core and forces developed depend on the fuel assembly configuration. Because the Mark-BW/MOX1 fuel assembly design uses the same configuration as other similar fuel types, the use of LYNXT for evaluating the fuel rod fretting and fuel assembly liftoff is acceptable.

Topical Report BAW-10186P-A, “Extended Burnup Evaluation,” contains a methodology that is approved for analyzing fuel rod bow and the acceptable limits on the amount of bow (Reference 18). The introduction of MOX will change the local power peaking which is an input to this method, therefore, the use of MOX fuel pellet material will not change the methodology. The acceptable limits are based on thermal-hydraulic considerations and as noted previously, the fuel assembly thermal-hydraulics will not change because of the use of MOX pellet material. Because the MOX fuel will not change the acceptable limits or the methodology, the use of the methodology for fuel rod bow contained in BAW-10186 is acceptable.

Topical Report BAW-10084P-A, “Program to Determine In-reactor Performance of BWFC Fuel Cladding Creep Collapse,” contains an approved methodology for evaluating material creep under irradiation conditions (Reference 20). The differences between MOX and LEU fuel are defined as part of the input to the COPERNIC computer code. Aside from the change in code input parameters, the methodology for calculating the cladding creep is independent of fuel pellet material. Because the parameters that are impacted by the use of MOX fuel are captured by the input to the COPERNIC computer code and the remainder of the methodology is unchanged, the use of the creep collapse methodology in BAW-10084 is acceptable for evaluating the behavior of MOX fuel.

Topical Report BAW-10199P-A, “The BWU Critical Heat Flux Correlations,” contains approved critical heat flux correlations (Reference 21). These critical heat flux correlations describe the heat transfer from the cladding material to the coolant; therefore, they are independent of the pellet material. The staff finds the use of these correlations for MOX fuel analysis acceptable because they are fuel pellet material independent.

Topical Report BAW-10133P-A, “Mark-C Fuel Assembly LOCA Seismic Analysis,” contains an approved methodology for calculating the fuel seismic response based on an elastic model for the cladding (Reference 25). This method is dependent on the fuel cladding material but is not fuel pellet material specific; therefore it is acceptable for use in analyzing MOX fuel seismic response.

4.0 CONCLUSION

The NRC staff reviewed the acceptance criteria and generic and proposed analysis methodology presented by Framatome in the topical report “MOX Fuel Design Report” and determined that the criteria and proposed analysis methods are performed in accordance with the guidance provided in SRP 4.2. The staff finds the criteria and proposed analysis methods outlined in this topical report acceptable based on the determinations provided in the evaluation section of this safety analysis report and concludes that the topical is acceptable for referencing by Duke for the fuel assembly design and testing plan of four Mark-BW/MOX1 LTAs.

This evaluation supports the conclusion that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be adverse to the common defense and security or to the health and safety of the public. Therefore, on the basis of the above review and justification, the staff concludes that the Framatome Advanced Mark-BW fuel assembly design is acceptable for use is Westinghouse 3 and 4 loop design reactors which use a 17 x 17 fuel rod array with LEU fuel subject to the conditions includes in this safety evaluation report.

5.0 CONDITIONS

The staff approves use of this topical report subject to the following conditions:

1) This fuel assembly design is approved for use with Mixed Oxide fuel.

2) The Mark-BW/MOX1 fuel assembly design is licensed for LTA use only to a maximum fuel rod burnup of 60,000 MWD/MT.

3) The Mark-BW/MOX1 Fuel assembly design may not be modified from the design presented in BAW-10238.

4) The gallium content of the plutonium must be limited to 120 ppb.

5) The Uranium oxide powder used in the MOX fuel pellets must be fabricated by using the ADU process.

6.0 REFERENCES

1. Letter from James Mallay, Framatome ANP to the USNRC, “Request for Approval of BAW-10238(P), Revision 1, MOX Fuel Design Report,” May 30, 2003.

2. Letter from James Mallay, Framatome ANP to the USNRC, "Partial Response to RAI on BAW-10238(P), Revision 1, MOX Fuel Design Report," October 8, 2003.

3. Letter from James Mallay, Framatome ANP to the USNRC, "Partial Response to RAI on BAW-10238(P), Revision 1, MOX Fuel Design Report," October 27, 2003.

4. Letter from James Mallay, Framatome ANP to the USNRC, "Partial Response to RAI on BAW-10238(P), Revision 1, MOX Fuel Design Report," November 24, 2003.

5. Letter from James Mallay, Framatome ANP to the USNRC, "Partial Response to RAI on BAW-10238(P), Revision 1, MOX Fuel Design Report," December 5, 2003.

6. Letter from James Mallay, Framatome ANP to the USNRC, "Partial Response to RAI on BAW-10238(P), Revision 1, MOX Fuel Design Report," December 16, 2003.

7. Letter from James Mallay, Framatome ANP to the USNRC, "this would be the letter for agreeing to the conditions," xxxxx, 2004.

8. BAW-10239-P-A, "COPERNIC Fuel Rod Design Computer Code," September 1999

9. SE on the Copernic code

10. DPC-NE-1005, Revision 0, Duke Power Company Nuclear Design Methodology Using CASMO-4/SIMULATE-3MOX, August 2001.

11. SE for Duke on Casmo/Simulate

12. DPC-NE-2005P-A, Revision 3, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, September 2002.

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14. BAW-10239P,Revision 0, "Advanced Mark-BW Fuel Assembly Mechanical Design Topical Report," April 2002.

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18. BAW-10186-P-A, Revison 1, Supplement 1, "Mark-BW Extended Burnup," November 2001.

19. BAW-10183-P-A, "Fuel Rod Gas Pressure Criterion (FRGPC)," July 1995.

20. BAW-10084-P-A, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," July 1995.

21. BAW-10199-P-A, "The BWU Critical Heat Flux Correlations," December 1994.

22. BAW-10199-P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," November 2000.

23. Letter from M.S. Tuckman, Duke Power to the USNRC, "Proposed Amendments to the Facitlity Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50," February 27, 2003.

24. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

25. BAW-10133-P-A, "Mark-C Fuel Assembly LOCA-Seismic Analyses," June 1986.

26. BAW-10133-P-A, Revision 1, Addendum 1, "Mark-C Fuel Assembly LOCA-Seismic Analyses," October 2000.

From: ELLIOTT Gayle F <Gayle.Elliott@framatome-anp.com>
To: "'Michelle Honcharik'" <MCH3@nrc.gov>
Date: 3/24/04 9:32AM
Subject: RE: Is there proprietary information in the faxed draft SE for BAW-10239

Michelle,

There is no proprietary information in the draft SEs for BAW-10238 or BAW-10239.

Gayle Elliott
Manager, Product Licensing
Framatome ANP - Regulatory Affairs
Office: (434) 832-3347
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----- Original Message-----
From: Michelle Honcharik [mailto:MCH3@nrc.gov]
Sent: Tuesday, March 23, 2004 3:50 PM
To: Gayle.Elliott@framatome-anp.com
Subject: Is there proprietary information in the faxed draft SE for BAW-10239

Gayle,
I need to put into ADAMS a copy of the draft SE for BAW-10239, that I faxed to you. Before I do so, I need to confirm that there is no proprietary information in the draft SE. Please reply via e-mail.
Thank you,

Michelle Honcharik
NRR/DLPM/PDIV-1
Project Manager
(301) 415-1774
O-7C16

ATTACHMENT 3