July 1, 2004

Mr. James F. Mallay
Director, Regulatory Affairs
Framatome ANP
3815 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR FRAMATOME ANP TOPICAL REPORT BAW-10238(P), REVISION 1, "MOX FUEL DESIGN REPORT" (TAC NO. MB7550)

Dear Mr. Mallay:

On May 30, 2003, Framatome ANP (FANP) submitted Topical Report (TR) BAW-10238(P), Revision 1, "MOX Fuel Design Report," to the staff. On March 31, 2004, an NRC draft safety evaluation (SE) regarding our approval of BAW-10238(P) was provided for your review and comments. By letter dated April 9, 2004, FANP commented on the draft SE. The staff's disposition of FANP's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The staff has found that BAW-10238(P), Revision 1, is acceptable for referencing by licensees for the fuel assembly design and testing plan of up to four Mark-BW/MOX1 lead test assemblies to the extent specified and under the limitations delineated in the TR and in the enclosed SE. The SE defines the basis for acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that FANP publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain historical review information, such as questions and accepted responses, draft SE comments, and original TR pages that were replaced. The accepted version shall include a "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, FANP and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Herbert N. Berkow, Director
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure: Safety Evaluation

July 1, 2004

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Sincerely,
/RA/
Herbert N. Berkow, Director
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Enclosure: Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

FRAMATOME ANP

PROJECT NO. 728

1.0 INTRODUCTION

By letter dated May 30, 2003, and its supplements dated October 8 and 27, November 24, December 5 and 16, 2003, and March 29, 2004, (References 1 through 7), Framatome ANP (FANP) requested approval of topical report (TR) BAW-10238(P), Revision 1, "MOX Fuel Design Report." Approval of the TR would permit the use of the fuel assembly mechanical design, Mark-BW/MOX1, for mixed oxide (MOX) lead test assembly (LTA) use. The FANP Mark-BW/MOX1 fuel assembly design is an evolution of the Mark-BW fuel design and includes new design features. This TR 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 evaluation contained in the subject TR demonstrated that the Mark-BW/MOX1 fuel design met the criteria defined in SRP Section 4.2.

In addition, FANP requested approval for making minor changes to the fuel assembly design that have been evaluated as described in this TR and for making the subsequent design changes without specific NRC review and approval. As indicated in the conditions, the Mark-BW/MOX1 fuel assembly design may not be modified from the design presented in BAW-10238 and BAW-10239. Additionally, FANP requested approval for using the LTAs as part of this TR review. This request is denied because approval for use of the LTAs is granted only by a plant-specific licensing action. 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.

FANP 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 considered in the context of a plant-specific licensing action. 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).

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

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

(1) This fuel assembly design is approved for use with MOX fuel.

(2) The Mark-BW/MOX1 fuel assembly design is approved for LTAs only to a maximum fuel rod burnup of 60,000 Megawatt-days/metric ton (MWD/MT).

(3) The Mark-BW/MOX1 fuel assembly design may not be modified from the design presented in BAW-10238(P), Revision 1 and BAW-10239.

(4) The gallium content of the plutonium dioxide (PuO_2) powder must be limited to 300 parts per billion (ppb). The 300 ppb limit includes the measurement uncertainty.

(5) The uranium dioxide (UO_2) powder used in the MOX fuel pellets must be fabricated by using the ammonium diuranate (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 specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 10 are met. Additionally, GDC 27 and 25 require that licensees maintain control rod insertability and core coolability. Loss-of-Coolant accident (LOCA) coolability requirements are contained in 10 CFR 50.46. The staff review process for new fuel designs is contained in SRP Section 4.2, “Fuel System Design.” The guidance provided within the SRP forms the basis of the staff’s review and ensures that the criteria of GDC 10 are met.

The regulation at 10 CFR 50.34, “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 for the review of TRs. As such, the staff review of this TR is 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, FANP considered four areas where the properties of MOX fuel differ from current low enriched uranium (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 UO_2 matrix. Depleted uranium contains about 0.25 percent U-235 with approximately 95 percent 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 (SE) related to Reference 8 and will not be repeated here (see Reference 9).

In current light water reactor (LWR) fuel, the fissionable component of the fuel is U-235. For MOX fuel, the fissionable components are Pu-239 and Pu-241. The isotopes Pu-239 and Pu-241 are present in LEU fuel, although in a different concentration than in MOX fuel. Therefore, some modifications of existing computer codes for the neutronic aspects of MOX fuel are necessary to properly evaluate a reactor core containing both MOX and LEU fuel. For the MOX LTAs, Duke 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 SE (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 UO_2 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 metallographic examination and/or autoradiography of fuel pellet samples from each batch of pellets. The fuel specification provides limits on the agglomerates. The U.S. fuel specification is consistent with the current European specification; therefore, FANP 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 core analyses: thermal-hydraulic and neutronic. The thermal-hydraulic mixed core analysis uses Duke-approved methods (References 12 and 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 SEs 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 of weapons-grade plutonium differs slightly from the isotope mixture of reactor-grade plutonium. Reactor-grade plutonium is derived from spent LEU fuel that is reprocessed after being discharged from a reactor 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 in 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 during the operating cycle. This difference in isotopes and their depletion with burnup has been modeled explicitly in the neutronics code.

For the MOX LTAs, Duke will use the approved CASMO-4/SIMULATE-3 MOX codes, 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 SE (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 300 ppb. This limit will be incorporated into the fuel specification.

The U.S. Department of Energy (DOE) Fissile Materials Disposition Program included testing of weapons-grade MOX fuel in the Advanced Test Reactor at the Idaho National Engineering and Environmental Laboratory. Two weapons-grade MOX fuel compositions were used. One composition was not treated to remove any of the gallium, resulting in a pellet average gallium content of 2.97 parts per million (ppm); while a second composition used a gallium removal step in the manufacturing process which resulted in pellets with gallium content of approximately 1.33 ppm. PIEs were performed after the rods were irradiated to 40 GWD/MT. These examinations demonstrated that no significant migration of the gallium occurred from the fuel to the cladding. Based on these tests, the staff believes that gallium will not migrate from the fuel to the cladding at an appreciable rate that will embrittle the cladding material. To confirm that gallium does not migrate with increased burnup, FANP has committed to provide the staff with the Oak Ridge National Laboratory MOX Test Fuel 50 GWD/MT PIE report. If the 50 GWD/MT PIE report shows evidence that gallium behavior changes appreciably with burnup, then the staff will review the issue. FANP also committed to include an analysis of the gallium content of the pellet and cladding in the hot cell PIEs planned for the twice and third burned fuel rods. The results of the irradiated gallium analysis will be compared to the gallium analysis performed on unirradiated archival MOX LTA fuel pellets and M5 cladding material taken from the same lot as the LTA cladding material.

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 TR. Therefore, nothing in this section is approved by this SE.

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 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 M5 material for the cladding, structural tubing, and grids (Reference 15).

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 UO_2 pellets. They are chamfered at the top and bottom to facilitate pellets loading into the rods and are dish shaped at the ends. This geometry 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 TR. The fuel pellet density is decreased from 96 percent theoretical density to 95 percent theoretical density. This change was made so that the theoretical density would be consistent with the MOX pellet density currently in 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 UO_2 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 TR. Based on the content of the TR, the staff concludes that a satisfactory description of the fuel assembly has been provided for this review.

In this TR, FANP 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 TR without modification.

The objectives of this fuel system safety review, as described in SRP Section 4.2, are to provide assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOO), (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 AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. The design bases for each criterion remains the same as defined in the Advanced Mark-BW fuel assembly (Reference 14).

3.5.1 <u>Fuel System Damage Criteria</u>

The design criteria relating to the fuel system damage should not be exceeded during normal operation including 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 Section 4.2 will be reviewed to confirm that the design criteria are not exceeded during normal operation for the Mark-BW/MOX1 design.

3.5.1.1 Stress

The design criteria for stress are 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). These design criteria are consistent with the acceptance criteria of SRP Section 4.2; therefore, the stress criteria are 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 TR. 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 TR. 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 criterion for strain is that the Mark-BW/MOX1 fuel rod transient strain (elastic plus plastic) limit should not exceed 1 percent for Conditions I and II events. This criterion is intended to preclude excessive cladding deformation during normal operation and AOOs. This design criterion is consistent with the acceptance criteria of SRP Section 4.2; therefore, this strain criterion 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 percent 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 criterion for cladding fatigue is that the Mark-BW/MOX1 maximum fuel rod fatigue usage factor shall not exceed 0.9. This design criterion is consistent with the acceptance criteria of SRP Section 4.2; therefore, this cladding fatigue criterion 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 40 years; therefore, the fuel rod will experience 20 percent 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 Section 4.2 guidance and the maximum fatigue is well below the design criteria limit, it has been demonstrated that the cladding fatigue acceptance criterion has been met.

3.5.1.4 Fretting

The design criteria for fretting are 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. These design criteria are consistent with the acceptance criteria of SRP Section 4.2; therefore, the fretting criteria are 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 performed well in the fretting arena. The Mark-BW/MOX1 and Advanced 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. Out of core testing on the Advanced Mark-BW fuel assembly design, including a 1000-hour endurance test, were also performed. 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 are that the Mark-BW/MOX1 fuel rod cladding best-estimate corrosion shall not exceed 100 microns. These criteria are intended to preclude potential fuel system damage mechanisms. The SRP does not provide 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. FANP 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, FANP has demonstrated that the oxidation, hydriding, and crud buildup for the Mark-BW/MOX1 fuel assembly design have met the acceptance criteria.

3.5.1.6 Fuel Rod Bow

The design criterion 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 for fuel rod bow specified in SRP Section 4.2, the SRP only requires that rod bow be included in the design analysis.

The FANP 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 are 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. These design criteria are consistent with the acceptance criteria of SRP Section 4.2; therefore, these axial growth criteria are 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 end of life (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 criterion 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 departure from nucleate boiling (DNB) propagation to occur. This design criterion was approved in Reference 19 and will continue to be valid since the parameters used in the methodology remain unchanged. Therefore, this criterion 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 are 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. These design criteria are consistent with the acceptance criteria of SRP Section 4.2, except for the exclusion of the pump overspeed transient. However, FANP has been previously approved to exclude this transient; therefore, the assembly liftoff criteria are acceptable for application to the Mark-BW/MOX1 fuel design.

FANP 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 Section 4.2 will be reviewed to confirm that the design criteria are not exceeded during normal operation and are properly accounted for during postulated accidents for the Mark-BW/MOX1 fuel assembly design.

3.5.2.1 Internal Hydriding

The design criteria for internal hydriding are 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. FANP maintains the fabrication level for total hydrogen in the fuel pellets to a level that is lower than the SRP Section 4.2 value of 2 ppm. These design criteria are consistent with the acceptance criteria of SRP Section 4.2 and are acceptable.

FANP 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 Section 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 criterion for cladding collapse is that the 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 FANP design criterion is consistent with the acceptance criteria of SRP Section 4.2, it is acceptable for application to the Mark-BW/MOX1 fuel assembly design.

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

FANP uses two approved critical heat flux (CHF) correlations in the analysis of DNB occurrence for the fuel (References 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 are acceptable for use to demonstrate that the design criteria are met during normal operation and AOOs.

3.5.2.4 Overheating of the Fuel Pellets

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

SRP Section 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. FANP 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 beginning of life 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 (PCI)

There are no generally applicable criteria for PCI failure in SRP Section 4.2. The two criteria that should be applied in accordance with SRP Section 4.2 are that the uniform strain of the cladding should not exceed 1 percent and fuel melting should be avoided. Since both of these criteria were addressed previously in this SE, the criteria for PCI are 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 emergency core cooling system (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°F on peak cladding temperature and 17 percent on maximum cladding oxidation must be met. FANP demonstrated through high temperature oxidation and quenching tests that the M5 cladding can meet these limits. The data and analysis to support this conclusion were 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 (Reference 24) recommends that the radially averaged deposition at the hottest axial location be limited to 280 cal/g. Reference 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. FANP developed new ballooning and flow blockage models for M5 cladding which were reviewed and approved in Reference 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 criterion 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 are divided into three categories:

- Operating Basis 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 insertability, and a coolable geometry within the deformation limits consistent with the 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 Section 4.2 guidance; therefore, they are acceptable for application to the Mark-BW/MOX1 fuel assembly design.

FANP 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, FANP 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 LTAs with UO_2 fuel in locations where the LTAs saw near-peak core power conditions. The LTAs 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. 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/MOX1 irradiation 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. pressurized water reactor (PWR) 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, FANP developed an LTA inspection and testing program. This program consists of poolside 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 two cycles of irradiation and also after the third irradiation cycle if an LTA is irradiated for a third 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 PIEs will be performed after the second irradiation cycle and the third irradiation cycle if an LTA is irradiated for a third cycle. 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 conditions. Therefore, the staff finds the planned minimum test matrix acceptable. The results of the second and third cycle PIEs, if an LTA is irradiated for a third cycle, will be necessary to support a future application for batch MOX fuel use.

3.8 Applicability of Previously Approved Methods to MOX Fuel

The 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.

TR 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 TR 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.

TR 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.

TR 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-10186P-A is acceptable.

TR 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-10084P-A is acceptable for evaluating the behavior of MOX fuel.

TR BAW-10199P-A, "The BWU Critical Heat Flux Correlations," contains approved CHF correlations (Reference 21). These CHF 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.

TR 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 CONDITIONS AND LIMITATIONS

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

(1) This fuel assembly design is approved for use with MOX fuel.

(2) The Mark-BW/MOX1 fuel assembly design is approved for LTAs 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(P), Revision 1 and BAW-10239.

(4) The gallium content of the PuO_2 powder must be limited to 300 ppb. The 300 ppb limit includes the measurement uncertainty.

(5) The UO_2 powder used in the MOX fuel pellets must be fabricated by using the ADU process.

5.0 CONCLUSION

The staff reviewed the acceptance criteria and generic and proposed analysis methodology presented by FANP in the TR "MOX Fuel Design Report" and determined that the criteria and proposed analysis methods are performed in accordance with the guidance provided in SRP Section 4.2. The staff finds the criteria and proposed analysis methods outlined in this TR acceptable based on the determinations provided in the evaluation section of this SE and concludes that the TR is acceptable for referencing by licensees for the fuel assembly design and testing plan of up to four Mark-BW/MOX1 LTAs.

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, “Acceptance of Draft Conditions for the SE on BAW-10238,” March 29, 2004.

8. BAW-10231-P, "COPERNIC Fuel Rod Design Computer Code," September 1999.

9. Safety Evaluation for Topical Report BAW-10231P, "COPERNIC Fuel Rod Design Code" Chapter 13, MOX Applications, January 14, 2004.

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

11. Draft Safety Evaluation for Duke Topical Report DPC-NE-1005P, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," February 20, 2004.

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

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

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

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

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

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

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 Facility 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," May 1974.

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.

Attachment: Resolution of Comments

Principal Contributor: U. Shoop

Date: July 1, 2004

RESOLUTION OF COMMENTS

ON DRAFT SAFETY EVALUATION FOR BAW-10238(P), REVISION 1,

"MOX FUEL DESIGN REPORT"

By letter dated April 9, 2004, Framatome ANP provided comments on the draft safety evaluation (SE) for BAW-10238(P), Revision 1, "MOX Fuel Design Report." The following is the staff's resolution of those comments.

1. FANP Comment: Delete the first two sentences of the second paragraph of the introduction. This statement is unnecessary since Condition 3 more clearly presents the NRC's intent.

   NRC Action: Leave the first sentence of the second paragraph as originally stated and replace the second sentence of the second paragraph with "As indicated in the conditions, the Mark-BW/MOX1 fuel assembly design may not be modified from the design presented in BAW-10238 and BAW-10239." The comment was not fully adopted in the final SE, because the new wording more clearly reflects the staff position and clarifies when a review is necessary.

2. FANP Comment: Delete "...both the poolside and hot cell..." from the last sentence of the third paragraph of the introduction. Since the draft SER is for lead test assemblies (LTAs) only, specific requirements for batch operation should be omitted.

   NRC Action: The comment was not adopted into the final SE, because the topical report requested LTA and batch approval. This part of the statement clarifies the staff expectations for a batch application and provides insight into what is missing from the original topical report.

3. FANP Comment: Change "acceptable" to "accepted," in the last line of the fourth paragraph in the introduction. Since the material has not been reviewed, there is no basis for concluding that the material is not acceptable.

   NRC Action: The comment was fully adopted into the final SE.

4. FANP Comment: Change "Pu-240" to "Pu-241," in the 4^{th} paragraph of Section 3.1.

   NRC Action: The comment was fully adopted into the final SE.

5. FANP Comment: Change "is" to "may be," in the third sentence of the fourth paragraph of Section 3.1. It is not valid to state that all codes require modifications for MOX fuel; it depends on the methodology and the desired level of accuracy.

   NRC Action: The comment was not adopted into the final SE, because the CASMO/SIMULATE suite was modified with the introduction of MOX fuel. Therefore, the use of the word "is" reflects what has been done.

6. FANP Comment: Change "... is use in Europe" to "... in use in Europe," in the third paragraph of Section 3.4.

   NRC Action: The comment was fully adopted into the final SE.

7. FANP Comment: Delete the last sentence of the first paragraph of Section 3.7.1, "Hot cell PIEs will be performed after 2 cycles of irradiation and after the third irradiation cycle." Replace this sentence with "The need to perform a third cycle hot cell PIE will be evaluated based on the direction of the DOE Fissile Materials Disposition Program." This is consistent with the wording in Section 3.2.2.

   NRC Action: The sentence was changed to read, "Hot cell PIEs will be performed after two cycles of irradiation and also after the third irradiation cycle if an LTA is irradiated for a third cycle." The comment was not adopted into the final SE, because the purpose of LTAs is to collect data (i.e., if hot cell information is not taken after the third cycle, what would be the point of irradiating the LTAs beyond the requested batch burnup?) If a third irradiation cycle is completed, then it is expected that the data will be collected.

8. FANP Comment: Delete the first sentence of the fourth paragraph of Section 3.7.1, "Hot cell examinations will be performed after the second and third irradiation cycles." Reword the second sentence to read from "For these tests..." to "For the hot cell examinations..." This is consistent with the wording in Section 3.2.2.

   NRC Action: The sentence was changed to read "Hot cell PIEs will be performed after the second irradiation cycle and the third irradiation cycle if an LTA is irradiated for a third cycle." The comment was not adopted into the final SE, because the purpose of LTAs is to collect data (i.e., If hot cell information is not taken after the third cycle, what would be the point of irradiating the LTAs beyond the requested batch burnup?) If a third irradiation cycle is completed, then it is expected that the data will be collected.

9. FANP Comment: Delete the last sentence of the last paragraph of Section 3.7.1, "The results of these PIEs will be necessary to support a future application for batch MOX fuel use." Since the draft SER is for LTAs only, specific requirements for batch operation should be omitted.

   NRC Action: The sentence was changed to read "The results of the second and third cycle PIEs, if an LTA is irradiated for a third cycle, will be necessary to support a future application for batch MOX fuel use." The comment was not adopted into the final SE, because the topical report requested LTA and batch approval. This part of the statement clarifies the staff expectations for a batch application and provides insight into what is missing from the original topical report.

Additional changes that were made:

The NRC staff deleted the last sentence of the last paragraph of Section 3.2.2, "The need to perform a third cycle hot cell PIE will be evaluated based on the direction of the DOE Fissile Materials Disposition Program." The statement did not correctly articulate the NRC staff position, which could be interpreted as inconsistent with later statements in the SE, as noted in FANP's comments to the draft SE.

In Section 3.5.1.8, Fuel Rod Internal Pressure, first paragraph, second to last sentence, "... it is fuel design independent," was replaced with "... the parameters used in the methodology remain unchanged."

In Sections 3.5.2 Fuel Rod Failure Criteria and 3.5.3.4 Fuel Assembly Structural Damage from External Forces, the reference to the Advanced Mark-BW design was changed to Mark-BW/MOX1 fuel assembly design, for correctness.

In Section 3.5.2.3, Overheating of Cladding, last sentence of last paragraph, was reworded as "Therefore, these correlations are acceptable for use to demonstrate that the design criteria are met during normal operation and AOOs."