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RULEMAKINGS AND
ADJUDICATIONS STAFF

NUCLEAR REGULATORY COMMISSION

Docket No. 50-413/414-OLA Official Exh. No. 38
In the matter of Duke Catawba
Staff ✓ IDENTIFIED 7/14/04
Applicant RECEIVED 7/14/04
Intervenor REJECTED
Cont'g Off'r
Contractor DATE
Other Witness
Reporter

April 5, 2004

Mr. H. B. Barron
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28202

SUBJECT: SAFETY EVALUATION FOR PROPOSED AMENDMENTS TO THE FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS TO ALLOW INSERTION OF MIXED OXIDE FUEL LEAD ASSEMBLIES (TAC NOS. MB7863, MB7864, MC0824, AND MC0825)

Dear Mr. Barron:

Enclosed is a copy of the Nuclear Regulatory Commission (NRC) staff's Safety Evaluation (SE) regarding your application submitted on February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3 and 4, December 10, 2003, and February 2 (two letters), March 1 (three letters), March 9 (two letters), March 16 (two letters), March 26 and March 31, 2004, to revise the Technical Specifications for the Catawba Nuclear Station to allow the use of four mixed oxide fuel lead test assemblies in one of the two Catawba units.

The issuance of this SE does not constitute NRC approval of your application to modify the licensing basis for the Catawba Nuclear Station. This SE documents the technical and regulatory disposition of the subject discussed within. NRC approval of your application, including its application for exemption from certain regulatory requirements, should it be appropriate, will be under separate correspondence. One or more supplements to this SE will be issued prior to or with the authorization of any change to the licensing basis for Catawba. These supplements will provide the publically available evaluation of security related issues and other matters as may be appropriate.

In the event of any comments or questions, please contact me at (301) 415-1493.

Sincerely,

/RA/

Robert E. Martin, Senior Project Manager
Project Directorate II-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: As stated

cc w/encl: See next page

NRC Staff Exhibit 1

Template=SECY-027

SECY-02

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RENEWED FACILITY OPERATING LICENSE NPF-35

AND

RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3 and 4, December 10, 2003, February 2, 2004, (two letters), March 1, 2004, (three letters), March 9, 2004, (two letters), March 16, 2004 (two letters), March 26 and March 31, 2004, Duke Energy Corporation, et al. (Duke, the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2 (Catawba), and to the McGuire Nuclear Station, Units 1 and 2 (McGuire), Technical Specifications (TS). The amendment request was revised by the licensee's letter dated September 23, 2003, to remove McGuire from the application. The licensee proposed to revise the TS to allow the use of up to four mixed oxide fuel (MOX) lead test assemblies (LTAs). Duke currently plans to load the four MOX LTAs into Catawba, Unit 1, in the spring of 2005. However, Duke has requested regulatory approval for both Catawba units to facilitate adjustments for changes in the LTA fabrication schedule, should any such changes occur. The supplemental letters provide additional clarifying information and did not expand the scope of the original Federal Register Notice (68 FR 44107, July 25, 2003).

This license amendment request is being made as part of the ongoing United States -- Russian Federation Fissile Material Disposition Program (FMDP). The goal of the FMDP is to dispose of surplus plutonium from nuclear weapons by converting the material into MOX fuel and using that fuel in nuclear reactors. In doing so, the plutonium will be rendered unsuitable for use in nuclear weapons and the increased radiation levels will reduce the threat of diversion of this material. Plutonium dioxide (PuO_2) powder supplied by the Department of Energy (DOE), will be blended with depleted uranium dioxide (UO_2) powder, and fabricated into MOX fuel pellets and MOX fuel assemblies. The four MOX LTAs will be loaded into Catawba instead of an equal number of low-enriched uranium (LEU) fuel assemblies for a minimum of two refueling cycles to be followed by post-irradiation examinations. These LTAs will be manufactured in France under the direction of Framatome Advanced Nuclear Power (ANP).

2.0 REACTOR SYSTEMS

This Safety Evaluation (SE) addresses the in-reactor performance and impact on the safety analyses for the MOX LTAs. The Nuclear Regulatory Commission (NRC) staff concludes that the MOX LTAs are capable of meeting the regulatory criteria addressed herein.

2.1 Regulatory Requirements

2.1.1 MOX Fuel LTAs Impact on Plant Operation

Fuel designs must ensure that the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 10, "Reactor Design," are met. Specifically, that appropriate margin be provided so that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Additionally, GDC 27, "Combined reactivity control system capability," and GDC 25, "Protection system requirements for reactivity control malfunctions," require that licensees maintain control rod insertability and core coolability. The NRC staff review process for new fuel designs is discussed in Standard Review Plan (SRP) 4.2, "Fuel System Design."

2.1.2 Loss-Of-Coolant Accident (LOCA) Safety Analysis

The requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," specify that each boiling or pressurized light-water cooled nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that the calculated cooling performance following a postulated LOCA conforms to the criteria contained within the rule.

The stated requirements can be met through an evaluation model for which an uncertainty analysis has been performed, as stated in 10 CFR 50.46:

> *(a)(1)(i)...the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. ...*
>
> *(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.*

Paragraph (b) of 10 CFR 50.46 specifies that: the calculated peak cladding temperature (PCT) shall not exceed 2200 degrees Farenheit (°F); the maximum cladding oxidation must not exceed 0.17 times the total cladding thickness before oxidation; the maximum hydrogen generation must not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel pellets were to react; the core must remain in a coolable geometry; and the core temperature shall be maintained at an acceptably low level and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

2.1.3 Non-LOCA Safety Analysis

According to 10 CFR Part 50, Section 34, "Contents of Applications; Technical Information," Safety Analysis Reports 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 must be submitted with an application. As part of the core reload process, licensees perform reload SEs to ensure that their safety analyses remain bounding for the design fuel cycle.

In addition, the licensee conducted all analyses using NRC approved codes and methods, resulting in conformance with GDC 11, "Reactor Inherent Protection," 10 CFR 50.46 (b) and other appropriate Updated Final Safety Analysis Report (UFSAR) Chapter 15 acceptance criteria. These acceptance criteria are addressed by the licensee in Tables Q12-1 through Q12-3 of the licensee's letter dated November 3, 2003.

2.1.4 Criticality Evaluation of MOX Fuel Storage in the Spent Fuel Pool (SFP)

Pursuant to 10 CFR Part 50, Appendix A, GDC 62, "Prevention of criticality in fuel storage and handling," the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes. The NRC staff reviewed the amendment request to ensure that the licensee will comply with GDC 62.

The regulatory requirement for maintaining subcritical conditions in SFPs is contained in 10 CFR Part 50, Section 50.68, "Criticality accident requirements." Since the licensee currently uses 10 CFR 50.68 as the licensing basis for its SFP, the NRC staff has reviewed the proposed changes against the appropriate parts of this section.

2.1.5 Technical Specification Changes

The regulations in 10 CFR 50.90 require a licensee to apply for an amendment to its license anytime a change to the TS is desired.

In an effort to reduce unnecessary changes to the TS not required by 10 CFR 50.36, "Technical Specifications," the NRC issued Generic Letter (GL) 88-16 (Reference 25), that provides guidance for relocating cycle-specific parameter limits from the TSs to a Core Operating Limits Report (COLR). This guidance allows a licensee to implement a COLR to include cycle-specific parameter limits that are established using an NRC approved methodology. The NRC approved analytical methods used to determine the COLR cycle-specific parameters are to be identified in the Administrative Controls section of the TSs.

2.2 Technical Evaluation

2.2.1 Description of MOX Fuel Lead Assembly Mechanical Design Features

The Mark-BW/MOX1 fuel assembly design is much the same as the Advanced Mark-BW fuel assembly design (Reference 26) 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 by the NRC's staff's SE in Reference 27, for use in Westinghouse three-and four-loop reactors that 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 28). The Mark-BW/MOX1 fuel assembly contains 264 fuel rods held in place by a structural cage of 11 spacer grids, 24 guide tubes, an instrument tube, and top and bottom nozzles. The MSMGs increase the flow turbulence along the hottest spans of the fuel rod. The intermediate spacer grids of the Mark-BW/MOX1 fuel assembly design are not mechanically attached to the guide thimble, instead they use ferrules around a third of the guide thimbles to limit the axial displacement of the intermediate grids. This allows the grids to float and reduces the axial forces on the guide thimbles and fuel rods.

2.2.2 MOX Fuel and Fuel Rod Design Features

The entire stack length of the Mark-BW/MOX1 fuel rod will be filled with MOX fuel pellets. 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 pellet loading into the rods and are dish shaped at the ends. This geometery configuration will reduce the tendency of the pellets to change into an hourglass shape under irradiation.

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 that the MOX fuel design topical report (Reference 29) addresses by stating that the axial gap between the top nozzle adapter plate and the fuel rods was analyzed to show that sufficient margin exists at the design rod average burnup to accommodate the fuel assembly growth and the fuel rod growth. 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 uranium oxide fuel rods. The lower burnup is consistent with current European burnup limits.

The isotopic mixture of weapons-grade plutonium differs slightly from the isotopic 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 in Reference 30 that considers the effect of the weapons-grade MOX fuel isotopes in performing the core neutronic calculations. The NRC staff approval of these codes is contained in the related NRC staff SE (Reference 31) and will not be repeated here.

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 have been investigated and models to predict the parameters have been developed and incorporated into the COPERNIC computer code (Reference 32). The NRC staff approval of this code is contained in the related NRC staff SE (Reference 33) and is not repeated here.

2.2.3 Design Evaluation

The fuel system design bases must reflect these four objectives as described in Section 4.2 of the SRP: 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. To satisfy these objectives, acceptance criteria are used for fuel system damage, fuel rod failure, and fuel coolability. The design bases and analyses demonstrating that the Mark-BW/MOX1 fuel design satisfies these objectives is contained in BAW-10238, MOX Fuel Design Report (Reference 29). The NRC staff approval of BAW-10238 is contained in the NRC staff's SE (Reference 34) and will not be repeated here.

2.3 Effects of MOX Fuel Lead Assemblies on Plant Operation

2.3.1 Nuclear Design

The primary active fuel material in MOX fuel is plutonium, which has different nuclear properties than conventional LEU fuel. However, even with these different nuclear properties, the impact of the four MOX LTAs will have an insignificant effect on core wide behavior. Core performance will be dominated by the nuclear properties of the remaining 189 fuel assemblies in the core. Duke performed a comparison of several key core wide physics parameters (critical boron concentration, control rod worths, moderator and fuel temperature coefficients) in a typical LEU core model that included the four MOX LTAs. The comparison showed that the physics parameters are very similar to those in a typical all-LEU core with no MOX LTAs (see Tables 3-7 through 3-10 of Reference 1).

The reload design process for a core with MOX fuel assemblies differs from the currently employed methods used with a LEU core, in the use of the NRC approved CASMO-4/SIMULATE-3 MOX code system (Reference 30), which is an update to the current CASMO-3/SIMULATE-3 code system. The licensee used the CASMO-4/SIMULATE-3 MOX codes to perform the required analyses of cycle-specific nuclear physics parameters and core transient behavior for both mixed LEU/MOX fuel cores and all-LEU fuel cores. Likewise, the licensee developed power distribution uncertainty factors that are used to evaluate predicted fuel performance with respect to established peaking limits by bench-marking the CASMO-4/SIMULATE-3 MOX code system against partial MOX fuel cores, all-LEU cores, and critical experiments. The licensee developed uncertainties for both LEU and MOX fuel assemblies. The detailed nuclear design methodology is described in Reference 30.

The presence of the Pu-239 in the MOX fuel impacts the fissionable isotopic contents of the MOX fuel. At the beginning of the cycle, the key difference between MOX fuel and LEU fuel is that Pu-239 is the predominant fissionable isotope in the MOX fuel. The substitution of a MOX fuel assembly for a LEU fuel assembly affects the assembly neutronic behavior, its neutronic interaction with the rest of the core, and the fission product concentrations. Neutronic interaction between MOX and LEU fuel assemblies arises through the energy spectrum of the neutron flux. The energy spectrum of the neutron flux for the MOX LTAs impacts the delayed neutron fraction (β_{eff}), the void reactivity effect, and the prompt neutron lifetime.

The fraction of delayed neutrons β_{eff} is lower in magnitude in MOX fuel than in LEU fuel. However, the use of four MOX LTAs will not result in any measurable decrease in the β_{eff} from a typical LEU core. Therefore, the NRC staff concludes that these slight differences in β_{eff} will not have any significant impact on plant operations.

During a LOCA, the effect of the coolant voiding as the system depressurizes is responsible for achieving reactor shutdown and maintaining low fission powers in the unquenched regions of the core. Figure 3-2 of the February 27, 2003, submittal provides a comparison of a void reactivity curve (effect on assembly k_∞) for a reference Framatome ANP designed LEU fuel assembly with a void reactivity curve calculated for a weapons grade MOX fuel assembly at the same conditions. A larger negative reactivity insertion occurs for the MOX fuel assembly than for the LEU assembly for all void fractions. This effectively suppresses the MOX fuel assembly power relative to the LEU assembly throughout a LOCA.

2.3.2 Thermal-Hydraulic and Mechanical Design

The MOX LTAs will reside within a core of Westinghouse LEU fuel assemblies. The LTAs will be surrounded by resident LEU fuel assemblies having the same physical dimensions and very similar hydraulic characteristics. The MOX LTA design employs MSMGs and the resident fuel design uses intermediate flow mixing grids. The design of these mixing grids is such that the pressure drop from the entrance to the MOX LTA to its exit is less than 4 percent lower than the pressure drop for a resident Westinghouse fuel assembly at design flow rates. Hence, flow diversion favoring one fuel assembly at the expense of the other design is expected to be inconsequential. Therefore, there will be no mixed core impact on the LOCA performance of the resident Westinghouse assemblies. As part of its normal core reload analysis process, prior to loading of the MOX LTAs into the core, the complete set of LTA LOCA calculations will be done with the average core modeled to simulate the hydraulic performance of the resident assemblies, providing a direct evaluation of the resident fuel effects on the MOX fuel lead assemblies.

2.4 Safety Analysis of MOX Fuel Lead Assemblies

2.4.1 Impact of MOX Fuel Lead Assemblies on LOCA Analysis

The fuel resident in the Catawba core prior to the insertion of the MOX LTAs is Westinghouse Robust Fuel Assembly (RFA) LEU fuel. The LOCA analysis of record for the LEU fuel is composed of large break and small break LOCA analyses. The large break LOCA was evaluated using the approved Westinghouse realistic methodology based on the WCOBRA/TRAC computer code (Reference 35). As part of the analysis of record a sensitivity study was performed to account for the mixed sources of the fuel assemblies. The licensee found the Mark-BW fuel to have an insignificant affect on the performance of the RFA fuel. The limiting case PCT was found to be 2056 °F at the 95^{th} percentile, and the maximum local oxidation was found to be 10 percent (Reference 1, Table 3-6).

The small break LOCA was evaluated using the approved Westinghouse NOTRUMP computer code (Reference 36). A mixed core penalty for the small break LOCA was assessed to be 10 °F, and was applied to the RFA fuel. The small break LOCA PCT was significantly lower than the large break LOCA PCT.

Evaluation of the LTA large break LOCA response was performed using the Framatome ANP approved Babcock & Wilcox Nuclear Technologies LOCA methodology for recirculating steam generator plants (Reference 37). This methodology conforms to the requirements of 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." Evaluation of the LTA performance under large break LOCA conditions found that the LTAs could experience a PCT of 2018 °F, and a maximum local oxidation of 4.5 percent. The lower results are due to placement of the assemblies in non-limiting core locations yielding a local peaking factor and linear heat generation rate below those of the resident fuel.

The impact of the MOX LTAs on the thermal-hydraulics for the LEU fuel currently residing in Catawba will be small since the MOX LTA fuel utilizes the Mark-BW/MOX1 design with intermediate mixing grids that has similar pressure drop characteristics to the existing LEU resident fuel.

The licensee reported the results of additional sensitivity studies on the effect of Plutonium loading and fuel-cladding gap factor in Reference 1, section 3.7. Ranging the plutonium loading from 2.3 percent to 4.4 percent affected the PCT by 1 °F, and doubling the gap size reduced the PCT by 13 °F, while increasing the maximum local oxidation by 0.1 percent. The reported analyses utilized the worst conditions from the sensitivity studies.

Based on the NRC staff review of the information provided, the NRC staff concludes that the effect of four MOX LTAs has been conservatively evaluated and has been demonstrated to be in compliance with the requirements of 10 CFR 50.46.

2.4.2 Impacts of MOX Fuel Lead Assemblies on Non-LOCA Analyses

The licensee considered the impact of the MOX LTAs on the non-LOCA UFSAR Chapter 15 events. The addition of four MOX LTAs to an otherwise all-LEU core will not have significant impact on the core average physics parameters shown in Tables 3-7 through 3-10 of the February 27, 2003, application for a typical Westinghouse pressurized-water reactor (PWR) core such as that at Catawba. The data presented in Tables 3-7 through 3-10 summarized the differences in various core physics parameters between two representative core models. One core model (designated MOX in the tables) had four MOX LTAs in locations typical of the planned LTA core. The second core model (designated LEU in the tables) had all LEU fuel assemblies. In the second core model the four MOX LTA locations were replaced with four LEU fuel assemblies that were chosen so that the boron letdown and assembly powers were as close as possible to the first core model with the four MOX LTAs. Depletion simulations were then run on both core models and the core physics parameters were calculated at various effective full power days during the simulation runs. The results of the simulated runs are presented in the referenced Tables 3-7 through 3-10, demonstrating that the presence of four MOX fuel assemblies in an otherwise all-LEU core does not produce a significant change in any of the core physics parameters.

Duke stated in the submittal that for the first cycle of operation, the four MOX LTAs will be placed in symmetric core locations that have no control rods in them. The planned core design is a checkerboard reload pattern similar to that used in previous cycles. The reload value for each physics parameter used in the safety analysis and maneuvering analysis will be confirmed to be within the reference values previously calculated as described in References 30 and 38 prior to core reload with the MOX LTAs consistent with normal licensee reload analysis processes. If any of the reload values fall outside the reference values, the core design or safety limits will be modified or changes made to the core operating limits as allowed in the COLR.

The licensee also addressed the transients and accidents that are sensitive to local physics parameters, such as: 1) control rod ejection, 2) rod cluster control assembly misoperation (withdrawal/drop), 3) steam system piping failure, and 4) fuel assembly misloading. A brief discussion of each scenario is presented below.

As stated above, during the first cycle of operation, the four MOX LTAs will be placed in symmetric locations in the core that through core loading design techniques do not require the LTAs to be controlled with a control rod, (referred to as unrodded locations). In addition, they will be located away from fuel assemblies having significant ejected control rod worth. This action is intended to reduce the impact of the power increase that would occur in a MOX LTA located in the vicinity of a rod ejection assembly. Also as alluded to above, maintaining key core parameters within present design limits insures that both core wide and localized responses to a rod ejection in a core with MOX LTAs are no more limiting than for a core containing only LEU fuel assemblies.

The licensee performed an analysis to determine energy generated in the assemblies adjacent to assemblies with an ejected rod. Specifically, the licensee performed a control rod ejection simulation with the four MOX LTAs placed in their most likely locations in a representative core. The licensee performed this analysis with the NRC approved SIMULATE-3K MOX code (Reference 30) and included appropriate conservatisms on ejected control rod worth, delayed neutron fraction, fuel temperature coefficient, moderator temperature coefficient, control rod trip worth, and trip delay time. The calculations showed that the peak enthalpy in the core at end of life, hot zero power conditions was 54 calories per gram and occurred in an LEU fuel assembly located face adjacent to the ejected control rod location. The peak enthalpy predicted in a MOX LTA was 30 calories per gram. Therefore, for the core design contemplated for the MOX

LTAs, the control rod ejection accident calculation results are lower than the current regulatory acceptance criteria.

The licensee also looked at a single control rod withdrawal and control rod drop events. These events are not expected to be impacted by the introduction of four MOX LTAs, because, as previously noted, the MOX LTAs will be in unrodded locations during in the first cycle of operation. For later cycles the assembly reactivity and rod worth for any control rod inserted in a MOX fuel assembly will be reduced to values that are below the limiting values, since these assemblies will be at least once or twice burned. Consequently, the MOX LTAs will not be placed in limiting core locations for a single rod withdrawal or drop. The reload values for the control rod worths will be maintained within the reference values contained in the safety analysis.

The licensee stated that, steam system piping failure with the most reactive rod stuck will not be impacted. The introduction of the four MOX LTAs in unrodded locations will not significantly alter the rod worth of the most reactive rod. The core reload design will control the worth of the most reactive rod and the target value for the reload will be less than the Westinghouse reload design values contained in the safety analysis for this accident.

The licensee considered operation with a misloaded fuel assembly. The NRC staff's conclusion is that administrative measures already in place for detecting misloaded assemblies, plus additional assurance provided by core power distribution measurements during plant startup, will provide ample assurance against misloaded fuel assemblies.

The administrative measures imposed by the licensee during core reloads are as equally effective for MOX fuel as they are for LEU fuel loads. By design, these MOX LTAs have a much lower thermal neutron flux than LEU fuel assemblies for the same power level. Therefore, a MOX fuel assembly misloaded into an LEU location (or vice versa) would be even more apparent from a core flux map than a misloaded LEU assembly in a LEU loaded core. In addition, the planned reactivity for the MOX fuel assemblies was chosen to be similar to the reactivity of the co-resident LEU assemblies. Accordingly, the equally reactive MOX LTAs would have no more of an impact if misloaded than a similar misloaded LEU fuel assembly. As a result, a misloaded MOX LTA would be readily detected, given that the incore detector signal for an LEU assembly loaded in a MOX LTA location would be much higher than the expected signal for the MOX LTA. As a result, core operation with a misloaded assembly will not be significantly impacted by the introduction of four MOX LTAs.

In summary, the NRC staff finds that, neutronically, all analyses were conducted using NRC approved codes and methods, resulting in conformance with GDC 11, 10 CFR 50.46 (b) and other appropriate UFSAR, Chapter 15 acceptance criteria, as provided in the response to requests for additional information (RAIs) 12-1 through 12-3, dated November 3, 2003 (Reference 12).

2.5 Criticality Evaluation of MOX Fuel Storage in the SFP

2.5.1 Background

Catawba, Units 1 and 2 SFPs each contain a single storage region with one storage rack design. All of the storage racks have the same cell center-to-center spacing (13.5 inches) and have no Boraflex neutron absorbing panels. Currently, LEU fuel assemblies are qualified as "Restricted," "Unrestricted," or "Filler," based on initial enrichment and burnup criteria. "Restricted" storage allows storage of higher reactivity fuel when limited to a specified storage configuration with lower reactivity fuel (filler assemblies). Using the same subcriticality requirements, which is a $K_{eff} \leq 0.95$, unborated, the criticality evaluation performed by the licensee for this submittal has determined an acceptable "Restricted" storage configuration for MOX LTAs in the Catawba SFPs. In this evaluation "Restricted" storage is allowed for MOX LTAs when limited to a specified storage configuration with lower reactivity LEU fuel.

The licensee evaluated the storage of MOX LTAs in the Catawba SFPs. Specifically, the analysis was performed to determine whether the current LEU fuel storage configurations and strategies employed at Catawba will be adequate to store MOX LTAs in accordance with regulatory subcriticality limits.

The typical layout of the two fuel buildings at Catawba is provided in Figure A3-1 of the February 27, 2003, submittal (Reference 1). Fresh fuel is first received in the new fuel receiving area and stored temporarily prior to being removed from its shipping container. Upon removal from the shipping container LEU fuel assemblies are placed in a new fuel storage vault (NFV) location for inspection and then are either kept in the NFV or transferred to the SFP for storage prior to reactor irradiation. MOX fuel assemblies, on the other hand, will be placed directly in the SFP once they have been received on-site. The NFVs will not be used to store MOX fuel assemblies. Fresh fuel and irradiated reload fuel assemblies (both LEU and MOX) are transported to the reactor via the water-filled Fuel Transfer Area. Discharged fuel assemblies from the reactor are also returned to the SFP through the Fuel Transfer Area.

The Catawba SFP contains just one storage region, that is, all rack cells are of the same design. The Catawba racks are arranged in a flux trap pattern, and the spacing between storage cells is sufficiently large enough (13.5 inches), and the cell walls are thick enough, as to not require Boraflex poison material to ensure sub-criticality.

The reference MOX LTA evaluated for SFP storage contains a total plutonium concentration of 4.37 weight percent up to a maximum fissile plutonium concentration of 4.15 weight percent and a maximum U-235 enrichment of 0.35 weight percent, as discussed in Section A 3.7 of the application. The Mark-BW/MOX1 fuel design parameters important to neutronic analysis (pellet diameter, fuel density, active stack length, rod pitch, etc.) are identical or nearly identical to those parameters of the current LEU fuel assemblies being used at Catawba. Table A3-2 of Reference 1 provides the plutonium and uranium nominal isotopic fractions for the unirradiated Mark-BW/MOX1 fuel. Expected manufacturing variations from the nominal values are also listed, and these variations are considered in the mechanical uncertainty analysis provided in the February 27, 2003, submittal.

2.5.2 Neutronic Behavior of MOX Fuel in the Catawba SFP

The MOX LTA's principal fissile material is Pu-239. Pu-239 is a more effective thermal and epithermal neutron absorber than U-235 (larger absorption cross-section). As a result, other thermal neutron absorbers in the MOX fuel lattice (such as boron) are worth less than in a LEU fuel lattice. The boron atoms, whether dissolved in the coolant or in lumped burnable poison rods, do not compete for thermal neutrons as effectively with the Pu-239 in MOX fuel as they do with U-235 in LEU fuel.

Another important effect is the reactivity characteristics of MOX fuel. Higher plutonium isotopes build up more quickly with burnup in MOX fuel than in LEU fuel, because the MOX fuel assemblies start with appreciable amounts of Pu-239. This difference in the buildup and burnup characteristics of plutonium isotopes results in a flatter MOX fuel reactivity curve (reactivity drops off less steeply with burnup) than an equivalent LEU fuel reactivity curve.

2.5.3 MOX Criticality Analyses

The NRC defined acceptable methodologies for performing SFP criticality analyses are provided in two documents:

(a) Proposed Revision 2 to Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis," (Reference 39) and

(b) Memorandum from L. Kopp (NRC) to T. Collins (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (Reference 40).

In determining the acceptability of the Catawba amendment request, the NRC staff reviewed three aspects of the licensee's analyses: 1) the computer codes employed, 2) the methodology used to calculate the reactivity, and 3) the storage configurations and limitations proposed. For each part of the review the NRC staff evaluated whether the licensee's analyses and methodologies provided reasonable assurance that adequate safety margins in accordance with NRC regulations were developed and could be maintained in the Catawba SFP.

2.5.4 Computer Codes

The SCALE 4.4 / KENO V.a computer code system (Reference 41) was employed by the licensee for analyzing the MOX and LEU fuel assembly criticality. This code system is the industry standard for conducting SFP criticality applications, and has been extensively benchmarked to both MOX fuel and LEU fuel critical experiments as well as reactor operational data (References 42-44).

As an added measure of conservatism, the licensee performed the criticality computations for this evaluation of the MOX LTAs considering only unirradiated MOX fuel. That is, no burnup credit was taken, and so no reactivity-equivalencing curves were necessary. KENO V.a does have the capability of modeling burned fuel. This requires first generating isotopic number densities, and then putting that isotopic data into KENO V.a. However, as noted above, this was not necessary for the MOX LTAs at Catawba.

The licensee reviewed several benchmark reports for using SCALE with MOX fuel that had been previously developed. References 42 through 44 describe the results from benchmarking SCALE against MOX fuel critical experiments and against isotopic measurements from reactor-irradiated (Beznau and San Onofre) MOX fuel. The benchmarking of SCALE 4.4 to MOX fuel critical experiments yielded good agreement in K_{eff} predictions, with similar biases and slightly higher uncertainties than those previously determined for LEU fuel. Additional critical experiments were reviewed and evaluated by the licensee in order to enhance the benchmarking effort. All of these MOX experiments contained a mixture of plutonium oxide and uranium oxide fuel with plutonium oxide concentrations ranging from 2.0 weight percent to 19.7 weight percent (References 45 through 48).

2.5.5 Methodology

The analyses conducted by the licensee for storing MOX and LEU fuel in the Catawba SFP storage racks were reviewed against the regulatory requirements of 10 CFR 50.68, "Criticality Accident Requirements." Given the above regulatory requirement, the MOX fuel criticality analysis for the Catawba SFP comprises the following general steps:

- The design information is obtained for the MOX LTAs and LEU fuel assemblies that are being or will be stored in the Catawba SFP. Design details for the SFP racks themselves are also necessary, in order to properly model fuel storage in these racks.
- SCALE 4.4 / KENO V.a computer models for the MOX LTA design and the highest-reactivity LEU fuel assembly design are constructed. These assemblies are modeled in the Catawba SFP storage racks.
- From these nominal models, mechanical uncertainties are determined.
- With the nominal models, K_{eff} results are determined for each MOX or MOX/LEU assembly configuration considered for that particular SFP storage rack. Various reactivity penalties are added to each K_{eff} result to account for mechanical uncertainties (from the previous step) and code methodology biases/uncertainties, which gives the no-boron 95/95 K_{eff} for that storage configuration combination.
- In the Catawba SFPs, the maximum calculated 95/95 K_{eff} results must be less than 0.95 for the no-boron cases.
- Several potential SFP accident scenarios are also evaluated, including an assembly misloading event, accidents that increase or decrease the fuel pool water temperature, and a heavy load drop (weir gate) event. The amount of soluble boron needed to keep the 95/95 K_{eff} at or below 0.95 is determined for each of these accidents, and the maximum amount required is verified to ensure it does not exceed the minimum SFP boron concentration for normal operations (2700 parts per million (ppm)) for Catawba.

In accordance with the guidance contained in References 39 and 40, the licensee performed criticality analyses of the Catawba, Units 1 and 2 SFPs. The licensee employed a methodology that combines a worst-case analysis based on the most reactive fuel type, and statistical 95/95 analysis techniques. The major components in this analysis were a calculated K_{eff} based on the limiting fuel assembly, SFP design and code biases, and a statistical sum of 95/95 uncertainties and worst-case delta-k manufacturing tolerances.

In performing the criticality analysis, the licensee first calculated a K_{eff} based on nominal core conditions using the SCALE 4.4 / KENO V.a code package. The licensee determined this K_{eff} from the limiting (highest reactivity) fuel assemblies stored in the SFP. The licensee performed its reactivity analyses for various enrichments, cooling times, plutonium concentration uncertainties, and the rack cell wall thickness. In performing these calculations, the licensee assumed appropriately conservative conditions such as assuming plutonium isotopic fractions of 94 percent Pu-239, 5 percent Pu-240, and 1 percent Pu 241. The exact plutonium isotopics of the MOX LTAs are expected to be similar to those presented in Table A3-2 of the February 27, 2003, submittal and, therefore, are less reactive than the assumed isotopics in the criticality calculations.

To calculate K_{eff}, the licensee added the methodology bias as well as a reactivity bias to account for the effect of the normal allowable range of SFP water temperatures. The licensee determined the methodology bias from the critical benchmark experiments. For each of the proposed storage configurations, the licensee analyzed the reactivity effects of the SFP water temperature. The licensee calculated the reactivity bias associated with a temperature decrease to the maximum density of water, 4 degrees Celsius (°C).

Finally, to determine the maximum K_{eff}, the licensee performed a statistical combination of the uncertainties and manufacturing tolerances. The uncertainties included the computer code system benchmarking biases and uncertainty, Plutonium concentration uncertainties, fuel density uncertainties, cell wall thickness uncertainties, center-to-center cell spacing uncertainties and mechanical uncertainties. The licensee determined these uncertainties to a 95/95 threshold that is consistent with the requirements of 10 CFR 50.68. By using the most limiting tolerance condition, (upper limit of 95/95), the licensee calculated the highest reactivity effect possible. This results in conservative margin since the tolerances will always bound the actual parameters. Once the reactivity effects for each of the tolerances were determined, the licensee statistically combined each of the manufacturing tolerances with the 95/95 uncertainties. The NRC staff reviewed the licensee's methodology for calculating the reactivity effects associated with uncertainties and manufacturing tolerances as well as the statistical methods used to combine these values.

For normal conditions in the Catawba SFPs, the maximum no-boron 95/95 K_{eff} in the MOX/LEU Restricted/Filler configuration remained below 0.95, specifically 0.9217.

For three of the accident conditions that needed to be evaluated for fuel storage (fuel assembly misload, dropped fuel assembly, and abnormal SFP temperature changes), no addition of boron was needed to maintain the 95/95 K_{eff} below 0.95 in accordance with 10 CFR 50.68.

The other accident condition considered by the licensee is the heavy load drop onto the SFP racks. The largest loads that can be carried over the Catawba SFPs are the weir gates (see their locations in Figure A3-1). These 3000 to 4000 pound steel gates, if dropped onto the SFP racks, are capable of crushing up to seven fuel assemblies. In accordance with NUREG-0612 (Reference 49), heavy load drop evaluations must assume the racks and the fuel assemblies within them are crushed uniformly to an optimum pin pitch. Figure A3-7 of Reference 1 depicts the model for this weir gate drop into the SFP. The affected assemblies are crushed into a tighter and tighter configuration until maximum reactivity is achieved. For the Catawba racks, this worst-case 95/95 K_{eff} (0.9429) still remained below the 0.95 limit, with 2700 ppm boron in the SFP. The NRC staff finds the licensee's methods conservative and acceptable.

2.5.6 Storage Rack Configurations

As mentioned in the appendix section A3.2 of the February 27, 2003, submittal, the Catawba SFPs contain one storage region. The licensee performed criticality calculations for various storage patterns in the Catawba SFP. Different types of rack storage patterns were presented in Figure A3-5 of the February 27, 2003 submittal. These racks will be used for storing MOX and MOX and LEU fuel in the Catawba SFP. The one region SFP is designated as "Restricted/Filler Storage." Fresh or irradiated MOX LTAs qualify as Restricted assemblies in these storage regions. In addition, LEU fuel assemblies that exceed their LEU Unrestricted enrichment limit or do not meet the minimum required burnup for LEU Unrestricted storage can be stored as Restricted fuel in these storage regions. Low-reactivity "Filler" fuel assemblies in this configuration must be LEU fuel assemblies that meet the Filler minimum burnup requirements provided in Table A3-5 of Reference 1. The NRC staff finds this designation to be acceptable based on its review of the licensee's submittal as described above.

2.5.7 SFP Storage Summary

In summary, the licensee has examined the feasibility of MOX fuel storage in the Catawba SFP. The reference MOX fuel design (the Mark-BW/MOX1) was identified and evaluated for storage in the Catawba SFP. The analytical methodology used included conservatisms such as neglecting axial leakage and taking no credit for burnup in MOX fuel. The results from all of these Catawba SFP criticality analyses demonstrate that a reference MOX fuel design, with a maximum fissile plutonium concentration of 4.15 weight percent, and a maximum U-235 enrichment of 0.35 weight percent, can be stored fresh or irradiated in the patterns shown in Figure A3-5, of Reference 1, without any modifications to the existing SFP storage racks. This evaluation is consistent with the planned lead assembly fuel design of 4.37 weight percent total plutonium and 0.25 weight percent U-235, demonstrating that it also can be safely stored in the SFP storage racks. The NRC staff has reviewed the licensee's evaluation and concludes that all regulatory criteria are met.

2.6 Technical Specification Changes

The use of MOX LTAs necessitates revising TS on spent fuel storage, design features, and administrative controls. The licensee submitted the proposed TS changes and technical justification for the changes in Reference 1.

TS 3.7.16 Spent Fuel Assembly Storage

Currently, the Catawba Limiting Condition for Operation (LCO) 3.7.16 specifies allowable LEU fuel storage configurations by reference to TS Table 3.7.16-1 and Figure 3.7.16-1. A revision to this LCO is proposed in this license amendment request to also allow storage of MOX LTAs as Restricted Fuel in the Catawba SFPs. The description of the Restricted Fuel classification is in Figure 3.7.16-1 that is revised to include MOX assemblies as qualifying fuel.

In addition, Surveillance Requirement (SR) 3.7.16.1 is revised since the current language refers to initial enrichment and burnup criteria, neither of which applies to MOX LTA storage. SR 3.7.16.1 currently reads: "Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with the specified configurations." The intent of SR 3.7.16.1 is to verify that a fuel assembly meets the necessary criteria for storage in the SFP. The proposed change is to delete the current wording and insert the same language as contained in McGuire SR 3.7.15.1, that reads: "Verify by administrative means the planned spent fuel pool location is acceptable for the fuel assembly being stored."

The proposed change applies equally to an LEU or MOX fuel assembly and still requires verification that any fuel assembly meets the appropriate storage requirements identified in the associated LCO prior to moving it into the SFP. The NRC staff finds the proposed administrative change to be acceptable.

TS 4.2.1 Fuel Assemblies

TS 4.2.1, Fuel Assemblies, currently specifies that each fuel assembly consist of a matrix of ZIRLO or Zircaloy fuel rods with an initial composition of uranium dioxide as feed material. A revision to add a sentence describing the MOX LTAs is proposed. The proposed sentence states: "A maximum of four lead assemblies containing mixed oxide fuel and M5™ cladding may be inserted into the Unit 1 or Unit 2 reactor core." The proposed change would incorporate the description of the LTAs into the TS. The NRC staff finds the proposed change to the TS 4.2.1 description of the fuel assemblies to be acceptable because the description is consistent with the licensing application design provided by the licensee that has also been reviewed and approved by the NRC staff.

TS 4.3.1 Criticality

The licensee proposed to revise the current language of TS 4.3.1.1 that provides a limit on the enrichment of LEU fuel that can be stored in the fuel racks, with language that provides enrichment limits on MOX fuel as well as LEU fuel. The NRC staff finds the proposed change to the TS 4.3.1.1 to be acceptable because the description is consistent with the licensing application design provided by the licensee that has also been reviewed and approved by the NRC staff.

TS 5.6.5 Core Operating Limits Report

In accordance with the guidance provided by the staff in GL 88-16 (Reference 25), Duke requested to add two approved methodologies to the list in the COLR section of the TS. The two methodologies include the "Duke Power Nuclear Design Methodology Using CASMO-4/ SIMULATE-3MOX" and "COPERNIC Fuel Rod Design Computer Code" methods of analysis.

The core neutronic parameters are evaluated using the approved "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3MOX" (Reference 30). These codes are approved for use in analyzing reactor cores that contain both LEU and four MOX LTAs. The NRC staff approval of these codes is contained in the related NRC staff SE (Reference 31) and will not be repeated here.

Fuel behavior is analyzed using the COPERNIC code. MOX parameters have been investigated through experimental results and models to predict these parameters have been developed and incorporated into the COPERNIC computer code, Reference 32. The NRC staff approval of the COPERNIC code with the MOX parameters is contained in the related staff NRC SE (Reference 33) and will not be repeated here.

Duke has identified the approved methodologies that are used to generate the cycle-specific parameters in accordance with GL 88-16. These methods are required for the analyses of the MOX LTAs in the Catawba core and have been approved by the NRC staff for MOX fuel analyses. Therefore, the NRC staff finds that adding these two methodologies to TS 5.6.5 - Core Operating Limits Report, is acceptable.

2.7 Reactor Systems Summary

The NRC staff reviewed the analysis methodology and supporting documentation presented by Duke in the licensing application and determined that the analysis methods are acceptable. The NRC staff finds the analysis in this licensing application to be acceptable based on the determinations provided in the evaluation section of this SE and concludes that associated modifications to the TS to implement the use of four MOX LTAs into one of the Catawba units are acceptable. The NRC staff's conclusion for the subjects addressed in this SE is based on a limitation of maximum fuel rod burnup to 60,000 MWD/MThm.

An LTA is designed to gather data on fuel performance. The LTAs are typically based on current production designs and are irradiated to obtain fuel performance data. In the past, as fuel performance data was obtained, it indicated that slight design modifications would be necessary. As a result, minor design changes have been implemented into the current production designs to retain high fuel reliability. Data from LTAs will also provide the basis for improved fuel designs and analytical models.

An LTA is a fuel assembly based on a currently available design. An LTAs' fuel cladding material is an NRC-approved cladding material. The assembly will receive pre-characterization prior to undergoing exposure in the "test" cycle that would permit the assembly to exceed the burnup limits of the COPERNIC fuel behavior code. The fuel assembly has been analyzed using currently approved fuel performance design models in COPERNIC and methods in BAW-10238 and demonstrated that the currently approved design limits are met for the extended burnup. Because the purpose of an LTA is to gather data on fuel performance including above approved burnup limits, the models and methods used for evaluation of the LTAs are not required to be approved to the projected burnups. The available data on MOX fuel performance above 50,000 MWD/MThm, while not statistically significant, indicates that the approved models can predict the fuel behavior and therefore are appropriate for use to this burnup so modifications to the approved models are not necessary. Use of the models above the approved burnup limit will only be used for analysis of the LTAs. Model performance will be shared with the NRC along with the PIE data results.

Pre-characterization measurements shall be assessed with the fuel performance design models and methods to ensure that the assembly will not exceed design limits after its cycle of exposure. Pre-characterization is the measurement of particular fuel performance parameters before the start of the cycle. Upon completion of the cycle of exposure, the LTA shall under-go a Post Irradiation Examination (PIE). Post Irradiation Examination of the LTA shall be documented in a PIE report and results of the PIE assessment shall be factored into future analysis to ensure that appropriate conservatisms are being maintained. In addition, tracking of the data results will provide the basis for developmental model creation to more accurately model fuel performance and to capture fuel performance fundamentals. Reports containing data gathered by the vendor/utility from the LTA program shall be presented to the NRC. Model performance shall also be tracked against data and presented to the NRC.

Because the fuel performance models are being extrapolated to burnups that have not been approved, the pre-characterization provides a measure of how much margin exists for a given design criterion to its limit, based on model predictions compared to the pre-characterization measurement. Comparison of pre and post cycle values, obtained from the PIEs, will yield the incremental effects that the cycle of exposure has on the LTAs. This provides a measure of whether an unknown phenomenon exists and is occurring in the LTAs. It also provides a very accurate measure of how well the predictive fuel performance models are behaving for the cycle of exposure.

3.0 DOSE CONSEQUENCES

3.1 Regulatory Evaluation

This SE section addresses the impact of the proposed changes on previously analyzed design basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The applicable regulatory requirements are the accident dose guidelines in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," as supplemented by accident-specific criteria in Section 15 of the SRP, the accident dose criteria in 10 CFR 50.67, "Accident Source Term," as supplemented in Regulatory Position 4.4 of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50, Appendix A, GDC 19, "Control Room," as supplemented by Section 6.4 of the SRP. Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in the following documents in performing this review.

- Safety Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors"
- Safety Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"
- RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
- SRP Section 15.0-1, "Radiological Consequence Analyses Using Alternative Source Term"
- SRP Section 15.1.5, "Steam System Piping Failures Inside and Outside Containment (PWR)," Appendix A
- SRP Section 15.3.3, "Reactor Coolant Pump Rotor Seizure"
- SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Appendix A
- SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment"
- SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Rupture (PWR)"
- SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," Appendix A and Appendix B
- SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents"
- NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook"

Since the guidance identified above was written for LEU fuel, the NRC staff considered appropriate changes related to the MOX fuel. These adjustments are addressed in this report.

The NRC staff also considered relevant information in the Catawba UFSAR, TSs, and several technical reports. The technical reports are listed as References 50 through 63.

3.2 Technical Evaluation

3.2.1 Background

The LEU fuel used in U. S. nuclear reactors consists of uranium oxides in which the concentration of U-235 is increased over that in the naturally occurring distribution of the uranium isotopes during manufacture, such that U-235 constitutes about 4 to 5 percent of the uranium by weight. In fresh LEU fuel, U-235 is the fissionable component. The concentration of U-235 is specified by the fuel designer and produced during the enrichment process. Prior to irradiation, LEU fuel has no significant plutonium concentration. During irradiation, however, U-238 absorbs neutrons and transmutes to the various isotopes of plutonium. Some of these plutonium isotopes are fissionable and add to the power output of the LEU fuel.

In the beginning of the U. S. nuclear reactor program, it was anticipated that the fuel cycle would be closed by reprocessing spent fuel to recover the usable plutonium and uranium for use as MOX fuel in reactors. In the case of MOX fuel, Pu-239 rather than U-235 provides most of the fissionable material. The plutonium obtained from reprocessing is blended with natural or depleted uranium during manufacture to obtain the plutonium concentration specified by the fuel designer. Demonstration projects conducted in the 1970's and 1980's resulted in the irradiation of MOX fuel assemblies at several U.S. power reactors including San Onofre, Ginna, Quad Cities, and Dresden. Similar efforts proceeded in foreign countries during this period. Domestic MOX research ended by 1980 as a result of a presidential executive order against reprocessing irradiated fuel. However, foreign programs continued and commercial MOX use is a reality in Japan, India, and a number of European countries today. As of the end of 2001, more than 30 thermal reactors worldwide use MOX fuel. Since the plutonium in this commercial MOX fuel was obtained from reprocessing spent reactor fuel, this fuel is known as reactor-grade MOX fuel. Since the plutonium in the proposed MOX LTAs is obtained from weapons material inventories, this fuel is known as weapons-grade MOX fuel.

Table 1: Nominal Unirradiated Fuel Isotopics, %

Isotope	U.S. LEU	European MOX	Proposed MOX LTA
wt% ^{234}U / U	0.03	----	----
wt% ^{235}U / U	3.2	0.24 - 0.72	≤0.35
wt% ^{236}U / U	0.02	----	----
wt% ^{238}U / U	96.75	92.77	95.28
wt% ^{238}Pu / Pu	----	0.88 - 2.40	≤0.05
wt% ^{239}Pu / Pu	----	53.8 - 68.2	90.0 - 95.0
wt% ^{240}Pu / Pu	----	22.3 - 27.3	5.0 - 9.0
wt% ^{241}Pu / Pu	----	5.38 - 9.66	≤1.0
wt% ^{242}Pu / u	----	2.85 - 7.59	≤0.1
wt% Pu / HM	----	4.0 - 9.0	4.37
wt%Fissile / HM	3.2	3.65 - 5.25	≤4.15

HM = Pu + U. May not sum to 100% due to rounding and ranges. Derived from data in licensee submittal, ORNL/TM-2003/2 [Ref.1], NUREG/CR-0200 V1 [Ref.2]

The two MOX fuel types differ in that the relative concentrations of plutonium and uranium and the distributions of their isotopes differ. Table 1 above compares the distribution of fissile and non-fissile isotopes in typical LEU fuel, typical commercial reactor-grade MOX fuel, and the proposed MOX LTAs. The differences in the initial fuel isotopics are potentially significant to accident radiological consequence analyses since the distribution of fission products created depends on the particular fissile material. If the fissile material is different, it follows that the distribution of fission products may be different. For example, one atom of I-131 is created in 2.86 percent of all U-235 fissions, whereas one atom of I-131 is created in 3.86 percent of all Pu-239 fissions. This is an illustrative example only in that the radionuclide inventory in the fuel at the end of core life depends on more than fission yield. Nonetheless, this shift in the fission product distribution needs to be evaluated for its impact on the previously calculated radiological consequences of DBAs.

The LEU fuel is enriched in the U-235 isotope, an operation that occurs on a molecular scale while the UO_2 fuel is in the gaseous phase. This processing results in fuel pellets with a high degree of homogeneity and uniform grain sizes. The proposed MOX LTA fuel will be manufactured in a process that involves blending of UO_2 and PuO_2 powders to achieve the desired Pu content. The MOX fuel pellets, therefore, are not as homogeneous as an LEU fuel pellet. This difference in pellet structure has the potential to affect the diffusion of fission gases through the fuel pellet and may impact the fraction of the pellet fission product inventory that is in the fuel rod gap between the pellet outer surface and fuel clad inner surface (i.e., gap fraction). It is generally understood that the fission gas release (FGR) rate for MOX fuel is greater than that for LEU fuel, given comparable enrichments and burnups. This behavior is primarily explained by the lower thermal conductivity of MOX fuel pellets that results in higher fuel temperatures than in LEU rods. Since the gap fractions are an input to the analyses of calculated doses from non-core melt DBAs, changes to the gap fractions associated with MOX fuel need to be considered.

In addition to the possible impact on gap fractions, the increased FGR has an impact on the fuel rod internal pressurization. The proposed MOX LTAs have fuel design features intended to compensate for the increased FGR. The decontamination of radioiodine released from fuel rods damaged during a design basis fuel handling accident (FHA) is a function of the rate of a bubble rising through the overlaying water in the SFP and the bubble size distribution that are functions of the fuel rod internal gas pressure. If it can be shown that the internal pressurization is unchanged or increased only slightly, then current analysis decontamination assumptions remain valid.

In summary, the NRC staff's review was focused on the potential impacts of the following three characteristics of weapons-grade MOX fuel:

(1) The fission product inventory in a MOX LTA is expected to be different from that of an LEU assembly due to the replacement of uranium by plutonium as the fissile material.

(2) The fraction of the fission product inventory in the gap region of a MOX LTA is greater due to the increased FGR associated with higher fuel pellet centerline temperatures of MOX fuel.

(3) The increased FGR can result in higher fuel rod pressurization.

The configuration of the MOX LTA is nearly identical to that of the LEU fuel assemblies currently in use at Catawba. There is no change in rated thermal power or any significant changes to other plant process parameters that are inputs to the radiological consequence analyses. As such, the only impacts on these analyses would be from changes in the fission product inventory and the gap fractions, and in the case of the FHA, changes in the SFP decontamination factor, if any.

In performing this review, the NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of DBAs, performed by Duke in support of its proposed license amendment. Information regarding these analyses are provided in Section 3.7.3 of Attachment 3 of the February 27, 2003, submittal and in supplemental letters dated November 3 and December 10, 2003, and February 2, March 1, and 16, 2004. The NRC staff reviewed the assumptions, inputs, and methods used by Duke to assess these impacts. The NRC staff performed independent calculations to confirm the conservatism of Duke's analyses. However, the findings of this SE are based on the descriptions of Duke's analyses and other supporting information submitted by Duke. Only docketed information, supplemented by technical information in reports identified in the references, was relied upon in making this safety finding.

3.2.2 Radiological Consequence Analyses

The radiological consequences of postulated accidents were discussed in Section 3.7.3 of the February 27, 2003, submittal. In its response dated November 3, 2003, to the NRC staff's RAIs, Duke provided supplemental information on the evaluation of the impact of MOX LTAs on DBAs. In this response, Duke stated that it had performed a combination of evaluations and analysis to assess the impact of the MOX LTAs. Duke described a process in which the various DBAs were categorized on the basis of how many fuel assemblies would be affected by that event. Duke identified two major categories:

(1) Accidents involving damage to a few fuel assemblies. These include FHAs and the weir gate drop (WGD) accident. A small number of assemblies are involved such that if the MOX LTAs were in the damaged population, as conservatively assumed, they would comprise all or a significant portion of the population.

(2) Accidents involving damage to a significant portion of the entire core. These accidents range from the locked rotor accident (LRA) with 11 percent core damage, the rod ejection accident (REA) with 50 percent core damage, to the large break LOCA with full core damage. In this case, the relative effect of damaging all four MOX LTAs is reduced as the fuel damage population increases. For example, in a DBA LOCA, all 193 fuel assemblies are postulated to be damaged. The four MOX LTAs constitute just 2 percent of all the fuel assemblies in the core.

To these categories, the NRC staff would add a third:

(3) Accidents whose source term assumptions are derived from reactor coolant system (RCS) radionuclide concentrations. These include, steam generator tube rupture, main steamline break, instrument line break, waste gas decay tank rupture, and liquid storage tank rupture.

The radionuclide releases resulting from these events are based on established administrative controls that are monitored by periodic surveillance requirements, for example: RCS and secondary plant specific activity LCOs, or offsite dose calculation manual effluent controls. Increases in specific activities due to MOX LTAs, if any, would be limited by these administrative controls. Since the analyses were based upon the numerical values of these controls, there can be no impact of MOX LTAs on the previously analyzed DBAs in this category.

3.2.3 MOX LTA Fission Product Inventory

Duke calculated the fission product inventory of the proposed MOX LTAs using the NRC-sponsored SCALE (Standardized Computer Analyses for Licensing Evaluation) system, version 4.4. SCALE is a multi-purpose computational system for analyses of nuclear facilities and spent fuel packaging. SCALE contains analytical modules that address topics such as radiation source terms and shielding, criticality safety, high-level waste classification, lattice physics, and heat transfer. SCALE was developed at the Oak Ridge National Laboratory (ORNL) for the NRC. It is currently maintained by ORNL under the co-sponsorship of NRC and DOE, under a software quality assurance (QA) program that includes configuration management, module and data revision control, documentation, verification and validation programmatic elements. SCALE module results have been benchmarked against actual measurements and against other domestic and international analytical capabilities.

Duke selected the SAS2H control module of SCALE for performing this work. SAS2H uses the point depletion code ORIGEN-S to compute time-dependent concentrations of a large number of nuclides. The nuclides are simultaneously generated or depleted through neutronic transmutation, fission, radioactive decay, input feed rates, and physical or chemical removal rates. ORIGEN-S is a variant of the ORIGEN (and ORIGEN2) code that was modified to replace the "pre-packaged" cross-section libraries with the ability to access a cross-section library created specifically for the problem defined by the user's input. The SAS2H control module processes the user's input, calls several modules to produce the ORIGEN-S data input and time-dependent cross-section libraries, and calls ORIGEN-S to perform the burnup and decay analysis. Because of this structure, SAS2H and ORIGEN-S calculations can be based on parameters that precisely match those of the specific problem being considered. This is a significant advantage for the present evaluation since it would address nuclides and reactions not included in pre-packaged LEU libraries.

Duke applied SAS2H to a series of cases structured to model combinations of accident sequence, MOX LTA plutonium concentrations, and LTA power histories. Duke states that the models were built including conservatisms. In particular, the NRC staff notes that Duke assumed that the plutonium concentration of the pins in the LTA was 5 percent. The nominal LTA fuel design calls for 176 fuel pins with a plutonium concentration of 4.94 percent; 76 pins at 3.35 percent, and 12 rods at 2.40 percent. The nominal average plutonium concentration is 4.37 percent. Conservatively basing the calculation on a 5 percent plutonium concentration provides margin to compensate for differences (e.g., manufacturing tolerances and power history differences) between the nominal design and the actual fuel as loaded in the core. Duke described the modeling of these variables in greater detail in its RAI response dated November 3, 2003. Duke also defined and analyzed an equivalent LEU assembly based on assembly burnup, LEU enrichment, and MOX fuel plutonium concentration.

The NRC staff reviewed Duke's use of the SCALE code, the SAS2H modules and the general approach taken. The NRC staff also reviewed the input values Duke used with SAS2H. The NRC staff finds SCALE, ORIGEN-2 and SAS2H to be appropriate analytical methodologies. The NRC staff also performed some confirmatory analyses and comparisons. First the NRC staff compared the Duke results to data derived from a report prepared by Sandia (Reference 52). The calculations described in that report were performed using ORIGEN2 with a PWR plutonium cross-section library. The NRC staff performed its own SAS2H analysis. Based on its review and confirmatory calculations, the NRC staff concluded that the Duke inventory analysis, as described in the docketed materials, used an appropriate analytical methodology and appropriate input parameters to assess the fission product inventory of a MOX LTA.

3.2.4 Impact on Gap Fractions

The Catawba licensing basis is in transition between the traditional TID-14844 (Reference 53) source term and the alternative source term (AST) from RG 1.183. Duke revised the Catawba licensing basis to selectively implement the AST for the fuel handling and WGD accidents by License Amendments Nos. 198 and 191 dated April 23, 2002, for Units 1 and 2 respectively. The licensing basis gap fractions for the FHA and WGD were those provided in Table 3 of RG 1.183. There are no licensing basis gap fractions for the DBA LOCA as TID-14844 assumes an immediate full core melt release. For the remaining accidents, the gap fractions are those specified in Safety Guide 25 and RG 1.77. Duke proposed a 50 percent increase in the current guidance on gap fractions to bound the expected increase due to the MOX LTAs. In support of the conclusion that this assumed increase would be bounding, Duke advanced an argument based on the work of an expert panel convened by the NRC to evaluate the applicability of the fission product release fractions specified in NUREG-1465 (Reference 54).

Duke stated that:

> Since [RG 1.183] Table 3 is based upon expert panel work which was published in [NUREG-1465] and the panel saw similarities in gap release rates between LEU and MOX fuel, it could be inferred that the gap release rates in [RG 1.183] Table 3 should also be valid for MOX fuel gap releases.

The NRC staff does not believe that this inference is adequate justification for assuming that the non-LOCA gap fractions in Table 3 of RG 1.183 would be applicable to the MOX LTAs as stated by Duke. The expert panel was not tasked to consider gap fractions for non-LOCA events. The panel's deliberations were limited to LOCAs and other severe accidents involving a significant portion of the core. Finding that a core wide average gap fraction might not change does necessarily support a conclusion that the gap fraction for the limiting fuel assembly has not been affected. Duke appears to challenge its own inference by noting in Response Q3(g) of the November 3, 2003, letter " . . . current data comparisons show fission gas release from MOX fuel pellets is generally greater than the fission product release from LEU fuel . . . "

Duke supplied a graph of measurements of FGRs from European (reactor grade) MOX fuel in its November 3, 2003 letter. By letter dated February 2, 2004, Duke provided an explanation of this graph. The majority of the plotted LEU data were obtained from 17 x 17 matrix fuel rods irradiated in Electricité de France (EdF) facilities. The MOX data were obtained from fuel rods irradiated in EdF PWRs operated in base-loaded or in load-following conditions. The fuel pellets were fabricated from depleted uranium and reactor-grade plutonium using the MIMAS process (Reference 55). The MOX fuel assemblies are radially zoned with typical plutonium concentrations ranging from 2 to 6 percent. The maximum axially-averaged linear heat generation rate (LHGR) during irradiation ranged from 4.7 kW/ft to 7.4 kW/ft. Following irradiation, the fuel rods were punctured and the gas collected and analyzed for helium, xenon, and krypton. With the exception of the origin of the feed plutonium, the irradiated fuel configuration and fabrication method of the MOX LTAs is closely comparable to the fuel assemblies that underwent the post-irradiation examinations for the EdF PWRs. This database is essentially the same as the fission gas data that was used to develop and qualify the COPERNIC FGR model. Although the maximum MOX LTA exposure will be 7.9 kW/ft (Reference 20, Table Q6-1), exceeding the range of the experimental data, this occurs only for a short period of time at the beginning of the cycle. Duke asserts that the FGR is generally insensitive to power peaking of this magnitude that occurs early in fuel lifetime. Given the relatively short decay half-lives of the more significant radionuclides, the NRC staff agrees.

The NRC staff had an independent analysis of FGR performed using the FRAPCON-3.2 computer code (Reference 56). Duke provided detailed fuel configuration data and projected power histories for the MOX LTAs. For this analysis, the FRAPCON-3.2 code was modified so that two FGR models were used. The primary model in FRAPCON-3.2 is the Massih model. The added model is based on the ANS-5.4 model that can predict the release of both stable noble gas elements and the radioisotopes. Although the Massih model is considered to be the more reliable model, it is only capable of predicting the release of stable noble gases. As such, both the ANS-5.4 and Massih models were run. The ANS-5.4 model calculation was structured to calculate the release values for the radioisotopes based on the Massih predictions for the stable isotopes. Consistent with ANS-5.4 recommendations, the diffusion coefficient for I-131 was assumed to be seven times that used for the noble gases and the diffusion coefficient for cesium isotopes was assumed to be two times that for the noble gases. Also consistent with ANS-5.4 recommendations, the release fractions for the longer-lived radionuclides Kr-85, Cs-134, and Cs-137 were calculated using the stable gas routine within the ANS-5.4 model and the diffusion coefficients identified above.

The accuracy of the FRAPCON-3.2 release fraction predictions is dependent on the data input for the analysis. It is particularly sensitive to the power history. Duke characterized the power history as being conservative and bounding for the expected MOX LTA power histories. The power history was tabulated as the fuel burnup at each time step and the radial peaking factor $F_{\Delta H}$ (F delta-H). From the values of time and burnup, the LHGR values for each time step can be calculated. The LHGR can also be calculated from the core average LHGR and the radial peaking factors. The two derived power histories are slightly different. As such, the release fraction calculation was performed for both power histories. Additionally, the average core power was increased by 5 percent to compensate for possible differences between the expected power history and the actual irradiation of the MOX LTAs. Peak gas releases and end-of-life gas releases were considered.

For a given power history, the uncertainty in the release fractions can be estimated based on the standard deviation of the FRAPCON-3.2 predictions of stable noble gases compared to the measured data from LEU and MOX fuel. The standard deviation for LEU fuel stable gas predictions is 0.026 absolute release fraction and the standard deviation for MOX fuel stable gas predictions is 0.048 absolute release fraction. The NRC staff has opted to use an overall standard deviation of 0.031 absolute release fraction for noble gases. The standard deviation for the radioisotopes was obtained by scaling the stable gas standard deviation by the ratio of the predicted nominal release of the radioisotope divided by the stable noble gas release value. The staff based this decision on the following considerations: (1) the mechanisms for release for LEU and MOX fuel is the same with the primary differences being the diffusion coefficients for MOX versus LEU–the uncertainties should be similar, and (2) the calculated value for MOX fuel is higher because of the limited number of MOX experimental data points that were considered compared to those considered for the LEU uncertainty.

Table 2: Release Fractions (Gap Fractions), in percent

	Kr-85	I-131	Other Noble Gases	Other Halogens	Alkali Metals
RG 1.183 Table 3	10.0	8.0	5.0	5.0	12.0
Duke Power Assumption	15.0	12.0	7.5	7.5	n/a
Staff Analysis EOC 3	13.5 (14.9)	0.2	0.1	n/a	17.7 (19.1)
Staff Analysis Peak Value	14.4 (16.8)	9.5 (10.5)	3.2 (3.5)	n/a	19.1 (21.6)

Table 2 shows the release fraction values obtained by this analysis. For comparison, the RG 1.183, Table 3 values and the release fractions assumed by Duke are tabulated. The bases of the NRC staff's values include the 5 percent power uncertainty factor discussed above and 2-sigma uncertainty adjustments. The peak values occur at the end of cycle two, corresponding to a projected burnup of about 47 GWD/MThm. The values in the parentheses are based on the power history derived from the $F_{\Delta H}$ values, as discussed above.

The only halogen considered by the ANS-5.4 model is I-131. The categorization of radioisotopes as noble gases, halogens, and alkali metals is on the basis of similarity in chemical behavior. The NRC staff believes that the chemical behavior of the iodine isotopes (and those of bromine) is sufficiently similar that the observed increase in the I-131 release fraction can be applied to the RG 1.183 "other halogens" value of 5.0 percent to obtain a value appropriate for the MOX LTAs. It is significant to note that the observed differences between the radionuclides Kr-85, Kr-87, Kr-88, Xe-133, and Xe-135 are correlated to the difference in the half-lives of these radionuclides. The iodine radioisotope half lives for I-132, I-133, I-134, and I-135, are much shorter than that for I-131. As such, the gap fractions for these radioisotopes would be less than that for I-131. Since the former radioisotopes are not significant contributors to dose, the NRC staff finds that the licensee's assumption of 7.5 percent as the gap fraction for the "other halogens" category is reasonable.

The NRC staff's peak values in Table 2 are bounded by the release fractions assumed by Duke with the following exceptions:

- Duke did not consider the increase in the gap fraction assigned to the "alkali metals" group. Duke stated that cesium need not be considered for the FHA and WGD in that the RG 1.183 acceptable assumptions for a FHA analysis provide that particulates, such as cesium, are retained by the SFP. The NRC staff agrees with this assessment. However, the NRC staff considered whether or not the significant increase in the gap fraction for cesium needed to be considered for the remaining accidents. At the present time, Duke has not been approved for use of an AST for DBAs other than the FHA and WGD. The current licensing basis analyses for the LOCA, LRA, and REA events are based on the TID-14844 source term that includes only noble gases and halogens. The increase in Cs-137 is not relevant to the current licensing basis at Catawba and is, therefore, not an issue for the present amendment request. The NRC staff notes that, if Duke should implement an AST at Catawba in the future, the gap fraction associated with Cs-137 will need to be explicitly addressed in the DBA analyses.

- The NRC staff analysis estimated a gap fraction for Kr-85 of 16.8 percent, which is an increase of 68 percent over the Kr-85 gap fraction for LEU and is greater than the 50 percent increase assumed by Duke. Given the relatively low significance of Kr-85 as a dose contributor in comparison to other radionuclides and the relatively small Kr-85 inventory in the core, the impact of this difference in the Kr-85 gap fraction on FHA and WGD postulated doses will be negligible. There is no impact on the comparative analysis of the LOCA, LRA, or REA events since Duke based this analysis on the difference in the I-131 inventories.

Based on the above, the NRC staff has determined that Duke's assumption of a 50 percent increase in the gap fractions in Table 3 of RG 1.183 is acceptable for the purposes of the present amendment request only, and should not be construed as a precedent for another licensing action at Catawba or any other reactor site. The gap fraction analyses are strongly dependent on the projected power history. If the actual power history is to deviate significantly from the projected power history, the gap fraction evaluation should be re-visited.

The NRC staff did not model reactivity insertion accidents in the FRAPCON-3.2 assessment of gap fractions. The NRC has a generic program plan for high-burnup fuel to address recent insights from reactivity insertion accident experiments performed on high burnup fuels. The criteria and analyses for reactivity accidents were identified for resolution (References 57 and 58). The issues identified in this program plan are generic to light-water power plants and fuel types and are, therefore, being resolved on a generic basis. They are not unique to MOX LTAs and need not be considered for the present amendment. The need for further regulatory actions, if any, will be determined based on the outcome of the program plan.

3.2.5 At-Power Core Damage Accidents

Duke considered the impact of the four MOX LTAs on the LOCA, LRA, and REA events. These DBAs were not explicitly re-calculated. Since the dose can be shown to be proportional to the fuel assembly inventory and gap fractions, Duke's approach to evaluating the potential impact of the MOX LTAs was to compare the relative differences in radionuclide inventory and determine a correction factor that could be applied to the results of the current analyses of record for these events. Duke used the MOX LTA and the equivalent LEU assembly source terms developed for the FHA and WGD accident re-analyses to perform this assessment. Duke selected the thyroid dose due to I-131 as the evaluation benchmark since the thyroid dose is typically more limiting than the whole body dose given the lesser margin between calculated thyroid doses and its associated dose criterion. Also, I-131 is generally the most significant contributor to thyroid dose due to its abundance and relatively long decay half-life. Duke determined that the I-131 inventory in a MOX LTA was 9 percent greater than that of an equivalent LEU fuel assembly. Since the observed increases in the other iodine isotopes were less than 9 percent, this factor could be conservatively applied to all iodines. Duke applied this 9 percent increase as a multiplier to the dose results in the current analyses of record for the LOCA, LRA, and REA events as discussed in its letter dated March 16, 2004. Duke also applied a correction factor of 1.5 to reflect the increased gap fractions associated with the MOX LTAs.

The current analyses of record assume that all fuel assemblies (193) are affected by a LOCA. For the LRA, 11 percent of the core (21 assemblies) are assumed to be affected; for the REA, 50 percent of the core (97 assemblies) are assumed to be affected. Duke assumes that the four MOX LTAs are in the affected fuel population replacing four LEU assemblies for each of these events. Duke's results are discussed below and are shown in Table 4 of this SE.

- For the LOCA, the four MOX LTAs represent only 2.1 percent of the 193 assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 1.32 percent. The thyroid dose increased to 90.2 rem at the exclusion area boundary (EAB), 25.3 rem at the low population zone (LPZ), and 5.37 rem at the control room. (Duke also applied this increase to the TEDE results from a LOCA analysis that was submitted as part of a separate proposed AST license amendment request that is still under review. Since a LOCA AST analysis is not part of the current licensing basis, and since the scaling did not consider the impact of the other nuclides that contribute to the TEDE, the staff did not rely on it in approving the MOX LTA amendment request.)

- For the LRA, the four MOX LTAs represent only 19 percent of the 21 affected assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 12 percent. The thyroid dose increased to 4.1 rem at the EAB, and 1.3 rem at the LPZ.

- For the REA, the four MOX LTAs represent only 4.1 percent of the affected 97 assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 2.63 percent. The thyroid dose increased to 1.03 rem at the EAB, and remained at 0.1 rem (increase masked by numeric rounding) at the LPZ.

Duke assessed the control room dose only for the LOCA since the control room doses from a LRA or REA are bounded by those for the LOCA. This is acceptable to the NRC staff.

A scaling approach is acceptable to the staff if the scaling represents the difference from the current licensing basis (LEU) to the proposed licensing basis (LEU plus MOX) and that the projected doses meet applicable acceptance criteria. In this case, Duke compared the I-131 inventory for a MOX LTA with that for an equivalent LEU assembly. Duke stated in its letter dated March 1, 2004, that the equivalent LEU source term was used in the interest of isolating the observed difference due to the difference in fuel isotopics between LEU fuel and the proposed MOX LTAs. Although the staff agrees that Duke's approach would isolate the differences, the staff believes that the before and after dose comparison is inconclusive since the "before" doses were not based on the equivalent LEU assembly. Duke also stated that the equivalent LEU assembly source term bounded the current licensing basis source term, which Duke characterized as a conservative situation. However, since the equivalent LEU assembly inventory appears in the denominator when calculating the dose multiplier, the dose result may not be conservative.

To address this, the staff performed an independent analysis. UFSAR Table 15-12 provides core inventory that is the basis of the current analyses of record source term. This table provides an I-131 core inventory of 8.9E7 Curies. This core inventory equates to an average fuel assembly I-131 inventory of 4.61E5 Curies. The MOX LTA I-131 inventory for a FHA or WGD analysis is 8.81E5 Curies. Dividing out the radial peaking factor of 1.65, to obtain a level comparison basis, yields an average MOX LTA I-131 inventory of 5.34E5 Curies. As such, the I-131 inventory in a MOX LTA is 15.8 percent greater than that used in the current analyses of record for the LOCA, LRA, and REA events.

- For the LOCA, the potential increase in the iodine release and the resulting thyroid dose is 1.53 percent. The thyroid doses increased to 90.4 rem at the EAB, 25.4 rem at the LPZ, and 5.38 rem at the control room.

- For the LRA, the potential increase in the iodine release and the resulting thyroid dose is 14.0 percent. The thyroid doses increased to 4.22 rem at the EAB, and 1.37 rem at the LPZ.

- For the REA, the potential increase in the iodine release and the resulting thyroid dose is 3.04 percent. The thyroid doses increased to 1.03 rem at the EAB, and 0.103 rem at the LPZ.

The NRC staff finds that the results of Duke's analysis for LOCA, LRA and REA events are acceptable in that the postulated accident doses will continue to meet applicable dose criteria. However, Duke's use of an equivalent LEU assembly is inappropriate. The NRC staff bases its finding on the minimal differences between the doses determined by Duke and those determined by the NRC staff, and on both sets of dose results meeting applicable dose criteria. This finding should not be construed as a precedent that Duke's comparative analysis approach will be found acceptable in another licensing action at Catawba or at any other reactor site.

The NRC staff considered the possible impact of radionuclides other than I-131 on the results obtained by Duke. Since the current analyses of record are based on the traditional TID14844 source term, only the krypton and xenon radionuclides need to be considered. The inventory of krypton isotopes in a MOX LTA is less than that in a corresponding LEU assembly. The inventory of some xenon isotopes in a MOX LTA increased between 7 to 11 percent with the exception of Xe-135, which increased by 189 percent. Using the MOX / LEU ratios (including a 1.5x gap fraction increase) developed above and conservatively considering only those noble gases that increased in concentration, the maximum increase in the whole body dose would be about 2 percent for the LOCA, 16 percent for the LRA, and 3.5 percent for the REA. The NRC staff found that the resulting whole body doses would remain within regulatory criteria.

3.2.6 Fuel Handling Accident and Weir Gate Drop Accident Radiological Consequences

Duke assessed the MOX LTA impact on doses for the FHA and WGD accidents by re-calculating the analyses of record with updated input data. Duke stated that with the exception of the fuel assembly isotopics the analysis models were basically the same as the FHA and WGD models described in Duke's license amendment request dated December 20, 2001. The staff reviewed those descriptions and approved that amendment request by letter dated April 23, 2002. That amendment selectively implemented the AST for the FHA and WGD at Catawba. Duke did revise the control room X/Q value for the unit vent releases from that approved in the earlier amendment. In lieu of assuming that the dual control room intakes have balanced flow rates, Duke assumes that 60 percent of the air being drawn into the control room is from a contaminated stream. This approach is consistent with the guidance of RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," and is, therefore, acceptable.

The results of these two re-analyses were tabulated in Tables Q3(b)-3 and Q3(b)-4 of Duke's submittal dated November 3, 2003, and are shown in Table 4 of this SE. Duke projected radiological consequences for the FHA of 2.3 rem TEDE at the EAB, 0.34 rem TEDE at the outer boundary of the LPZ and 2.1 rem TEDE in the control room, increases of about 64 percent over the previous analysis for LEU fuel. Duke projected radiological consequences for the WGD of 3.5 rem TEDE at the EAB, 0.5 rem TEDE at the outer boundary of the LPZ, and 3.3 rem TEDE in the control room, increases of about 58 percent over the previous analysis for LEU fuel. These results remain within applicable regulatory limits.

As noted above, the FGR for MOX fuel is greater than that for LEU fuel. The increase in FGR will result in increased fuel rod pressure. The scavenging of released radioiodine by the SFP water is a function of the bubble transit time through the overlying pool water which, in turn, is a function of fuel rod pressurization. The acceptable effective pool decontamination factor of 200 is derived from data from tests involving fuel rods pressurized to no more than 1200 psig. Duke stated that its analyses of the internal rod pressure would remain below the 1300 psig that Duke states is their criteria. Duke uses a Westinghouse methodology to justify the acceptability of the 1300 psig pin pressure. The NRC has not endorsed the cited Westinghouse topical report as a generically acceptable methodology. As an adjunct to the gap fraction analyses, the NRC staff had an analysis performed of the fuel rod pressure. This analysis was based on a power history derived from the $F_{\Delta H}$ and core average LHGR data docketed by Duke, increased by 5 percent, with a two-sigma uncertainty added to the nominal FGR. This analysis showed that the rod pressure would be 1105 psia, which is less than the 1200 psig specified in Safety Guide 25. Based on the NRC staff's analysis, Duke's use of an effective pool decontamination factor of 200 continues to be acceptable.

The NRC staff performed confirmatory analyses of the FHA and the WGD. The NRC staff used a MOX LTA source term generated using the SCALE SAS2H computer code. For the FHA, this source term was decayed for 72 hours and multiplied by the radial peaking factor of 1.65. For the WGD, the NRC staff used the inventory from four MOX LTAs and three equivalent LEU fuel assemblies, decayed for 19.5 days, and multiplied by the radial peaking factor of 1.65. The results of the NRC staff's analyses confirmed the results obtained by Duke. Details on the assumptions found acceptable to the NRC staff are presented in Table 3 of this SE. The doses estimated by the licensee for the postulated FHA and WGD (See Table 4) were a small fraction, as defined in RG 1.183, of the 10 CFR 50.67 dose criteria and are, therefore, acceptable.

3.2.7 Fresh MOX LTA Drop

This accident analysis is not currently part of the Catawba licensing basis. Duke performed this analysis to assess the radiological consequences of a drop of a fresh MOX LTA prior to it being placed in the SFP. Duke correctly stated that plutonium isotopes have a much higher specific activity than uranium isotopes and could present a more severe radiological hazard if inhaled. Although the configuration of the MOX pellets and LTA fuel rods provides protection against inhalation hazards, it is conceivable that some plutonium might become airborne if the MOX LTA is severely damaged. Duke's analysis of this event was performed to be applicable to both McGuire Nuclear Station and Catawba Nuclear Station using values chosen to bound the parameters at either station.

Duke described the analysis as involving assumptions and methodologies that were used in the calculations supporting the MOX Fuel Fabrication Facility (FFF) construction authorization request. The NRC staff reviewing the FFF reviewed those calculations and found them to be consistent with NRC staff guidance in NUREG/CR-6410 and, therefore, acceptable. The review for the case of dropped fuel within the FFF was documented in the MOX FFF draft SE report dated April 30, 2003. Note that the specific case addressed in the present licensing action, dropped fuel in a reactor fuel building environment, was not considered during the MOX FFF review. The NRC staff has not previously used the guidance of NUREG/CR-6410 for DBA analyses for power reactors.

The following analysis description is from NUREG/CR-6410, revised to reflect the parameter values for the present application. The release of the radioactive material is found from the expression below:

$Q = MAR \times DR \times ARF \times RF \times LPF$

Where:

Q is the quantity of material that enters the environment, in kilograms

MAR is the quantity of the material at risk and is the kilograms of the uranium and plutonium isotopes in the fuel assembly that is postulated to be dropped. Duke documented the isotopic breakdown in Table Q3(a)-2 in the November 3, 2003, RAI response.

DR is the damage ratio of the material actually impacted by the event. Duke assumed that 1 percent of all the fuel pellets in the dropped fuel assembly are damaged from the fall. Duke states that the value is applicable to drops from heights up to 30 feet.

ARF is the atmospheric release fraction which is the fraction of the impacted material that can actually become airborne. Duke calculated a value of 1.96E-4 from curve fits to experimental data observed in a study performed by Sandia National Laboratory (Reference 59).

RF is the respirable fraction of the released material. Duke assumed that all of the material in the release was respirable.

LPF was defined as the fraction of airborne material that breaches the containment barrier. For the current application, this parameter is used for the fraction of airborne material not removed by the filters. Duke assumes credit for only one filter bank in the flow path from the SFP to the atmosphere and the control room and credits a filter efficiency of 95 percent.

The dose consequence of the release is found by the following expression:

$D = Q \times \chi/Q \times BR \times DCF \times Sp.A$

Where:

D is the dose. Although the FFF implementation called for the committed effective dose equivalent (CEDE), Duke opted to report the results in terms of TEDE. This is effectively equivalent in that TEDE is the sum of the CEDE and the deep dose equivalent, the latter being negligible in an accident involving a fresh fuel assembly.

Q is the release quantity solved above.

χ/Q is the atmospheric dispersion coefficient. Duke used a bounding value of 9.0E-4 sec/m^3. The Catawba value would have been 5.5E-4 sec/m^3. A bounding value of 1.74E-3 sec/m^3 was used for the control room.

BR is the breathing rate taken as 3.47E-4 m^3/sec. This value is consistent with regulatory guidance.

DCF is the dose conversion factor, rem/µCi

Sp.A is the specific activity of the plutonium or uranium isotope, µCi/kg. For its confirmatory calculations, the staff used the following relationship to determine the specific activity:

$$Sp.A = \lambda N = \frac{0.693}{T_{1/2}} \cdot \frac{m}{A} \cdot N_a$$

Where N_a = 6.025E23, m is 1 gram, A = atomic weight.

As noted, the overall methodology was previously accepted by the NRC staff reviewing the FFF and has been determined to be appropriate for the present application. Only two input values need to be considered further. The first, DR, was taken as 0.01. The FFF draft SE report found that the DR for pellets exposed to overpressurization gas flows and pressurized rods that are breached are 0.01 and 0.001 respectively. The NRC staff also considered the analysis of a dropped fuel assembly in Section 3 of Sandia Report SAND87-7082 (Reference 59). This evaluation postulated a 30-foot drop of a typical Westinghouse 17 x 17 irradiated fuel assembly. The evaluation concluded that the drop would result in fracturing the bottom 1.3 inches of the fuel pellets. The radial expansion of the fuel pellets causes the fuel rod clad to fail. Less than 1 percent (0.01) of the pellets are affected. The NRC staff notes that the experimental data were obtained with irradiated fuel. The physical properties of the fuel pellets and cladding (e.g., brittleness) are less limiting for fresh fuel. Based on these considerations, Duke's assumed value of 0.01 is acceptable for the present application.

The value of ARF is the fraction of particles released from the damaged pellets. The value of ARF derived by Duke was 1.96E-4. In developing this value, Duke used the methodology of Section 3.3.4.8 of NUREG/CR-6410. The method is based on the observation that when a hard, cohesive brittle solid material is impacted and crushed by some force (usually another solid) the first solid absorbs some or all of the impacting kinetic energy and can form fine particles. However, the impact might not make all the released material airborne. Following the guidance of NUREG/CR-6410, Duke used the Argonne National Laboratory data correlation shown below to arrive at the ARF value of 1.96E-4:

$$ARF = 3.27E\text{-}11 \times E^{1.131}$$

Where, E is the energy density in Joules per cubic meter. The energy density is the product of the drop height, the pellet density, and the gravitational constant.

The correlation assumes that the pellet is struck by a solid of equal or greater cross-sectional area in a free fall. In the actual case, the pellets in the lower portions of the fuel rods are compressed by the deceleration of the fuel assembly as the fuel rods impact the lower nozzle as the assembly strikes the floor. A portion of the energy exerted on the fuel pellets is dissipated in pellet-clad interaction and in the expansion of the fuel pellet causing bulging and rupture of the lower fuel rod cladding. Duke stated that by not considering the fuel assembly its analysis is conservative. Duke based this conclusion on the fact that including the fuel assembly structural materials would reduce the energy density projected for the drop, and that the analysis ignored the reduction in the energy density that would result from dissipation of momentum forces by pellet-clad interactions and the deformation of fuel assembly components. Based upon these considerations, the NRC staff has concluded that Duke's value for ARF is adequately conservative and consistent with the deterministic nature of this analysis.

Details on the assumptions found acceptable to the NRC staff are presented in Table 3. The EAB and control room TEDE estimated by the licensee for the postulated fresh fuel assembly drop were less than 0.3 rem. This is a small fraction of the 10 CFR 50.67 dose criteria and is, therefore, acceptable.

3.2.8 Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by Duke to assess the radiological impacts of operation with four MOX LTAs at either Catawba unit. With the exception of deviations that are identified and dispositioned above, the NRC staff finds that Duke used analysis methods and assumptions that are consistent with the conservative regulatory requirements and guidance identified in Section 3.1 above. The NRC staff compared the doses estimated by Duke to the applicable criteria identified in Section 3.1. Based on its review as documented above, the NRC staff finds that the licensee's conclusion that the EAB, LPZ, and control room doses from postulated design basis accidents will continue to meet the acceptance criteria identified in Section 3.1 is acceptable. Therefore, the NRC staff concludes that use of four MOX LTAs at either Catawba unit is acceptable with regard to the radiological consequences of postulated design basis accidents.

3.3 Spent Fuel Pool Cooling

10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel storage and handling and radioactivity control," requires the SFP to be designed with provisions for decay heat removal. Using SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," and the further updated guidance developed during extended power uprate reviews (Reference 84), the NRC staff reviewed the effect on the SFP cooling capability of adding four MOX fuel assemblies to the SFP. As shown in Figure 3-12 of Attachment 3 to the licensee's letter, dated February 27, 2003, the MOX fuel has a decay heat about 2 percent higher than that of the regularly used LEU fuel seven days (168 hours) after shutdown of the reactor, which is about the time of peak SFP temperature. Since the four MOX LTAs are only a small fraction of the fuel transferred to the SFP during refueling, the change in decay heat represents a negligible change in the total decay heat of all fuel stored in the pool. The effect of the relatively higher decay heat of the MOX fuel on the SFP cooling system will diminish with time because the decay of the fuel will lower the decay heat. Therefore, the NRC staff determined that placing four MOX fuel assemblies in the SFP will have a negligible effect on SFP cooling capability.

TABLE 3
ANALYSIS ASSUMPTIONS

Source Term

Core power (includes 2% uncertainty penalty), MWt	3479
Specific power level, MW/assembly *(Includes 1.65 peaking)*	29.745
Fuel assemblies	193
Fuel pins per assembly	264
Fuel pellet temperature, °K	1085.34
Cycle burnup, MWD/MThm	
MOX	16,950
LEU	60,000
Heavy metal per assembly, MThm	0.4626
Plutonium concentration wt% Pu / hm in MOX assembly	5.0
Uranium enrichment, wr% U / U in LEU assembly	4.0
Fuel isotopics	
^{238}Pu / Pu wt%	0.025
^{239}Pu / Pu wt%	92.5
^{240}Pu / Pu wt%	6.925
^{241}Pu / Pu wt%	0.5
^{242}Pu / Pu wt%	0.05
^{234}U / U wt%	0.0017
^{235}U / U wt%	0.25
^{236}U / U wt%	0.0012
^{238}U / U wt%	99.7471
Fuel clad	
Density, gm/cc	6.50
Temperature, °K	656.1
Zirconium, wt%	98.873
Nobium, wt%	1.0
Oxygen-16, wt%	0.127
Moderator	
Density, gm/cc	0.711
Boron, ppm (cycle average)	900
Temperature, °K	580.43
Model	
Lattice	squarepitch
Fuel pin pitch, cm	1.2598
Outside diameter of fuel in pin, cm	0.8191
Clad outside diameter, cm	0.95

Clad inside diameter, cm	0.8357

Fuel Handling Accident and Weir Gate Drop

Radial peaking factor	1.65
Number of damaged fuel assemblies	
FHA	1
Weir gate drop	7
Decay time, days	
FHA	3
Weir gate drop	19.5
Fuel rod gap fractions	
I-131	0.12
Kr-85	0.15
All other noble gases, iodines	0.10
Alkali metals	0.0
Iodine species fractions	
Elemental	0.9985
Organic	0.0015
Particulates	none
Water depth, ft	23
Pool scrubbing factor, effective	200
Release modeling	
Immediate release from fuel through pool to building / CNMT	
100% release from building / CNMT within 2 hours	
No credit for building holdup or filtration prior to release	
Control Room Volume, ft^3	117,920
CRAVS start delay time, minutes	30
Unfiltered inleakage, cfm	
Before CRAVS start	2100
After CRAVS start	100
CRAVS filter flow, cfm	
Recirculation	1500
Outside air makeup	2000
Total	3500
CRAVS filter efficiency, %	
Elemental iodine	99
Organic iodine	95
Control room occupancy factors	
0-24 hr	1.0
24-96 hr	0.6
96-720 hr	0.4
Control room breathing rate, m^3/s	3.47E-4

Offsite breathing rate, m^3/s	
0-8 hrs	3.47E-4
Atmospheric dispersion factors, s/m^3	
EAB 0-2 hours	4.78E-4
LPZ 0-8 hours	6.85E-5
Control Room 0-2 hours	1.04E-3

Drop of a Fresh MOX LTA

Number of dropped assemblies	1
Fraction of pellets in assembly that are affected	0.01
MOX LTA loading, kg U +Pu	462.6
Composition, %	
Pu	5
U	95
Weight percent	
wt% Pu-238 / Pu	0.025
wt% Pu-239 / Pu	92.50
wt% Pu-240 / Pu	6.925
wt% Pu-241 / Pu	0.50
wt% Pu-242 / Pu	0.05
wt% U-234 / U	0.0017
wt% U-235 / U	0.25
wt% U-236 / U	0.0012
wt% U-238 / U	99.747
Height of drop, ft	23
Airborne Respirable Fraction	1.0
Fraction of damaged pellet that becomes airborne	1.96E-4
Filter efficiency, %	95
Control room occupancy factors	
0-24 hr	1.0
24-96 hr	0.6
96-720 hr	0.4
Control room breathing rate, m^3/s	3.47E-4
Offsite breathing rate, m^3/s	
0-8 hrs	3.47E-4
Atmospheric dispersion factors, s/m^3	
EAB 0-2 hours	9.00E-4
Control Room 0-2 hours	1.74E-3

TABLE 4
DESIGN BASIS ACCIDENT DOSES BY LICENSEE

	All LEU Core	LEU Core Plus 4-MOX LTAs	Acceptance Criteria
LOCA, *rem Thyroid*			
EAB	89	90.2	300
LPZ	25	25.3	300
Control Room	5.3	5.37	30
Locked Rotor Accident, *rem Thyroid*			
EAB	3.7	4.14	30
LPZ	1.2	1.35	30
Rod Ejection Accident, *rem Thyroid*			
EAB	1.0	1.03	75
LPZ	0.1	0.103	75
Weir Gate Drop Accident, *rem TEDE*			
EAB	2.2	3.5	6.3
LPZ	0.31	0.5	6.3
Control Room	2.1	3.3	5.0
Fuel Handling Accident, *rem TEDE*			
EAB	1.4	2.3	6.3
LPZ	0.21	0.34	6.3
Control Room	1.3	2.1	5.0
Fresh LTA Drop, *rem TEDE*			
EAB	n/a	<0.3	2.5
Control Room	n/a	<0.3	5.0

3.4 Reactor Vessel Materials

Section 3.6.1 of the Technical Justification in the licensee's application dated February 27, 2003, contains an evaluation of the impact of using four MOX LTAs on the integrity of the reactor vessels in Catawba, Units 1 and 2. In a letter dated February 2, 2004, the licensee provided additional information that evaluated the impact of four MOX LTAs on the reactor vessel surveillance program.

3.4.1 Regulatory Evaluation

The NRC staff has established requirements in 10 CFR Part 50, Appendices G, "Fracture Toughness Requirements" and H, "Reactor Vessel Material Surveillance Program Requirements," (10 CFR Part 50, Appendices G and H) and 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," (PTS rule) to protect the integrity of the reactor vessel in nuclear power plants. 10 CFR Part 50, Appendix G requires the pressure-temperature (P-T) limits for an operating plant to be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Appendix G to the ASME Code) were applied. The impact of radiation embrittlement on P-T limits is determined using the methodology in RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." 10 CFR Part 50, Appendix G also requires the Charpy upper-shelf energy (C_vUSE) to be greater than 50 foot-pounds (ft-lbs) throughout the operating life of the reactor vessel unless lower values can be justified. 10 CFR Part 50, Appendix H requires nuclear power plants to establish a reactor vessel surveillance program to monitor changes in the fracture toughness in the reactor vessel beltline materials. 10 CFR 50.61 provides the fracture toughness requirements for protecting the reactor vessels of pressurized water reactors against the consequences of pressurized thermal shock (PTS). 10 CFR 50.61 requires licensees to perform an assessment of the reactor vessel materials' projected values of the PTS reference temperature, (RT_{PTS}), through the end of their operating license.

3.4.2 Technical Evaluation

3.4.2.1 Pressurized Thermal Shock

The PTS rule requires each licensee to calculate the RT_{PTS} value for each material located within the beltline of the reactor pressure vessel at the expiration of its license. The RT_{PTS} value for each beltline material is the sum of the unirradiated nil ductility reference temperature (RT_{NDT}) value, a shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation of the material (i.e., ΔRT_{NDT} value), and an additional margin value to account for uncertainties (i.e., M value). 10 CFR 50.61 also provides screening criteria against which the calculated RT_{PTS} values are to be evaluated. For reactor vessel beltline base-metal materials (forging or plate materials) and longitudinal (axial) weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 270 °F; for reactor vessel beltline circumferential weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 300 °F. RG 1.99, Revision 2, provides an expanded discussion regarding the calculations of the shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation and the margin value to account for uncertainties. In this RG, the shift in the RT_{NDT} value caused by exposure to high energy neutron irradiation is the product of a chemistry factor and a fluence factor. The fluence factor is dependent upon the neutron fluence. The chemistry factor is dependent upon the amount of copper and nickel in the material. Since the amount of copper and nickel in the material does not change with MOX fuel, the only factor to be evaluated in determining the impact of radiation on reactor pressure vessel embrittlement is the neutron fluence.

In Reference 65 the NRC staff evaluated the protection from PTS for the Catawba, Units 1 and 2 reactor vessels. For Catawba, Unit 1, the NRC staff calculated an RT_{PTS} value for the limiting beltline material at the end of the extended operating term (60 years of operation) of 62 °F. For Catawba, Unit 2, the NRC staff calculated an RT_{PTS} value for the limiting beltline material at the end of the extended operating term of 133 °F. The neutron fluence at the end of the extended license term for Catawba, Units 1 and 2 are 3.12×10^{19} neutrons/square centimeter (n/cm^2) and 3.16×10^{19} n/cm^2, respectively. In order for these materials to reach the PTS screening criteria, the neutron fluence would have to increase more than ten times the value at the end of the extended operating term.

The only factor in the RT_{PTS} calculation affected by the MOX fuel is the neutron fluence. In a letter dated February 2, 2004, the licensee explained in Attachment 3 why the use of four MOX LTAs has a negligible impact on neutron fluence as follows:

> The use of four MOX fuel lead assemblies will have no significant impact on the end-of-life fluence experienced by a Catawba reactor vessel. While the neutron energy spectrum from plutonium fissions is slightly higher than the spectrum from uranium fissions, the four MOX fuel lead assemblies represent only about 2 percent of the 193 fuel assemblies in the core.
>
> Duke plans to use the MOX fuel lead assemblies for three operating cycles. For the first two cycles, the MOX assemblies will be loaded in the interior of the core (e.g., core location C8). For the third cycle, one or more MOX fuel lead assemblies will most likely be loaded in a core location at or near the core periphery (e.g., core location C14). A representative core loading map for the first cycle is shown in Figure Q11-1 of [the licensee's letter dated October 3, 2003]. It should be noted that the actual MOX fuel assembly core locations have not been finalized and will be determined as part of the cycle specific reload design. As discussed below, the incremental impact of the four MOX fuel lead assemblies on reactor vessel fluence will be insignificant.
>
> In [the licensee's letter dated October 3, 2003], response to Question 11, Duke showed that using four MOX fuel assemblies during the first cycle of operation will have a negligible impact on the fast flux in the core. At the beginning of the first cycle, Figure Q11-2 of [the licensee's letter dated October 3, 2003] shows that the maximum calculated impact is a fast flux increase of 6.4 percent in the MOX fuel location itself (C8). Peripheral core locations are the most important with respect to the leakage of neutrons out of the core, and the maximum increase in fast flux in a peripheral fuel assembly is only 1.6 percent at the beginning of the first cycle. The small incremental impact of using MOX fuel on fast flux decreases further with burnup, because conventional LEU fuels assemblies produce more and more of their power from plutonium fissions as their burnup increases. Figure Q11-3 shows that at the end of the first cycle the impact of using four MOX fuel lead assemblies on the fast flux is less than 1 percent in all core locations.

> Burnup effects will make the incremental impact of using MOX fuel during the second cycle even smaller than during the first cycle. In the third cycle, with MOX fuel loaded in an exterior core location, any MOX fuel-related increase in the fast flux would have more potential to affect the fluence at the vessel. However, the difference between a twice-burned MOX fuel assembly and a twice-burned LEU fuel assembly is very small. As noted in Reference [66], at a burnup of 50 gigawatt-day per ton, "...only 36 percent of LEU fuel fissions are in uranium, so most of the power is coming from plutonium fissions. At this burnup the characteristics of LEU fuel have become very similar to those of MOX fuel." Accordingly, during the third cycle of irradiation there will be little difference between the neutron energy from a MOX fuel assembly and the neutron energy from a twice-burned LEU fuel assembly that would otherwise be loaded at the expected location on the core periphery. Therefore, the impact of four MOX fuel lead assemblies on vessel fluence should be negligible during all three cycles of operation.

The NRC staff concludes that using the lead MOX fuel assemblies as described by the licensee will have a negligible impact on the neutron fluence and the RT_{PTS} value. Since the neutron fluence would have to increase by more than ten times the value at the end of the extended period to reach the PTS screening criteria, the staff concludes that the Catawba, Units 1 and 2 reactor vessels will have adequate fracture toughness for protection against PTS while using MOX LTAs and the PTS analysis in Reference 65 will not be affected by the use of the MOX fuel lead assemblies.

3.4.2.2 P-T Limits and C_vUSE

P-T limits increase and C_vUSE decreases as neutron fluence increases. P-T limits were reviewed by the NRC staff in Reference 65. The NRC staff's evaluation of the P-T limits concluded that the limits satisfy the requirements in Appendix G to Section XI of the ASME Code, and Appendix G of 10 CFR Part 50 and that the licensee used the methodology in RG 1.99, Revision 2, for determining the impact of neutron radiation on the beltline materials. In Reference 65, the NRC staff concluded that the Catawba, Units 1 and 2 reactor vessels will have C_vUSE greater than 50 ft-lbs throughout the period of extended operation. Since the increase in neutron fluence using the MOX LTAs is negligible, as discussed in the previous section, this will have no impact on reactor vessel embrittlement, P-T limits and C_vUSE.

3.4.2.3 Reactor Vessel Surveillance Program

In References 64 and 65, the NRC staff reviewed the surveillance capsule withdrawal schedule for Catawba, Units 1 and 2. The NRC staff concluded that the surveillance program was being implemented in accordance with Appendix H of 10 CFR Part 50 and that the capsule withdrawal schedule for Catawba, Units 1 and 2 was acceptable. There will be no surveillance capsules in the Catawba, Units 1 and 2 reactor vessels during the use of MOX LTAs. However, an ex-vessel cavity dosimetry program is being implemented at both Catawba units. This program will supplement the surveillance capsule program and monitor the reactor vessel fluence. Ex-vessel dosimetry was installed in Catawba, Unit 1 in 2003, and will be installed in Catawba, Unit 2 in 2004. The ex-vessel cavity dosimetry program will confirm that the predictions of vessel fluence used to assess vessel embrittlement are conservative.

Since MOX LTAs will have a negligible impact on the neutron fluence received by the reactor vessel, as discussed above in Section 3.4.2.1, no change in the reactor vessel surveillance program is necessary.

3.4.2.4 Summary

Based on the NRC staff's review and evaluation of MOX LTAs, the NRC staff has determined that for Catawba, Units 1 and 2, the reactor vessel RT_{PTS} values will be less than the screening criteria in 10 CFR 50.61, the reactor vessel surveillance program, P-T limits, and C_vUSE will not be affected by the use of MOX LTAs. On the basis of the above regulatory and technical evaluations of the licensee's justifications for TS changes, the NRC staff concludes that the licensee's proposed TS changes are acceptable.

3.5 Occupational Dose, Routine Effluents

3.5.1 Regulatory Evaluation

The focus of the NRC staff's evaluation in Section 3.5 of this SE is with respect to whether the proposed changes to the TS are consistent with requirements of 10 CFR Part 20 and the criteria of Appendix I to 10 CFR Part 50 in the areas of occupational and public dose. The regulatory requirements and guidance on which the NRC staff based its acceptance are as follows:

Regulations

- 10 CFR 20.1101, "Radiation protection programs."

- 10 CFR 20.1201, "Occupational dose limits for adults."

- 10 CFR 20.1301, "Dose limits for individual members of the public."

- 10 CFR 50.34a, "Design objectives for equipment to control releases of radioactive material in effluents - nuclear power reactors."

Guidance

- 10 CFR Part 50, Appendix I, "Numerical guides for design objectives and limiting conditions for operation to meet the criterion "As Low As is Reasonably Achievable" for radioactive material in light-water-cooled nuclear power reactor effluents."

3.5.2 Technical Evaluation

This evaluation is on Sections 5.6.1 and 5.6.2 of the licensee's application, "Plant Effluents" and "Impacts to Human Health" respectively.

In Section 5.6.1, "Plant Effluents," the licensee has evaluated the overall impact that the proposed use of MOX LTAs would have on its radioactive gaseous and liquid effluent releases. The licensee concluded that there will be no anticipated changes in the type or amount of radiological effluents resulting from the use of MOX LTAs from that of its current LEU fuel. The licensee states that it will continue to maintain its radioactive gaseous and liquid effluents within license conditions and regulatory limits.

The licensee's conclusion is based on its evaluation of the similarity of MOX fuel to the current LEU fuel, both from a fuel design and fission product inventory perspective, and on the limit of having only four out of 193 fuel assemblies containing MOX fuel.

The licensee evaluated the types and amount of fission products available for release in effluents. As fuel is irradiated, both activation and fission products are created. The activation products are created in the reactor coolant and fission products are produced inside the fuel rods. Activation products that are created are a function of impurities and the chemistry of the reactor coolant and the thermal neutron flux that the materials encounter. Thermal flux is significantly lower in MOX fuel than in LEU fuel, which would tend to reduce the level of activation products. However, for four lead assemblies this is expected to be an insignificant effect.

Fission product inventories and fuel gap inventories in particular are of the same order or magnitude in both MOX fuel and LEU fuels. In particular, the amount of iodine and noble gas that would be released into the reactor coolant in the event of a leaking fuel rod would be similar. Additionally, any liquid or gaseous effluents would be processed by the plant liquid waste and waste gas systems prior to release into the environment. These waste treatment systems would limit radioactive discharges to the environment through the use of hold-for-decay, filtering, and demineralization. The licensee states that the plant treatment systems are capable of treating these radioactive effluents since the types of radioactive material in MOX and LEU fuel are the same and the curie content of MOX fuel is of the same order of magnitude as LEU fuel. Thus, the licensee is expected to maintain the same level of radioactive control and remain within regulatory limits with the MOX fuel as has been maintained with the LEU fuel.

In Section 5.6.2 of its application, "Impacts to Human Health," the licensee has evaluated the overall impact that the proposed modification would have on its workers (occupational exposure) and to members of the public.

For occupationally exposed workers, the licensee estimates that there will be slight increases in radiation exposure during the handling of MOX fuel during receipt and handling operations. The increase in dose is due to a higher dose rate from a fresh MOX LTA as compared to a fresh LEU fuel assembly. The total neutron and gamma dose rate at 10 centimeters from the face of a fresh MOX LTA averages about 6 mrem/hour, falling off to about 1.8 mrem/hour at 100 centimeters. This is a relatively low radiation field; however, it is larger than that associated with a LEU fuel assembly, which has virtually no radiation field at these distances. The initial receipt and handling activities for one MOX LTA could result in a conservatively estimated total occupational dose in the range of 0.020 to 0.042 person-rem. However, the licensee will use the application of the as low as reasonably achievable principle to try to effect lower doses than are estimated. Radiation doses of this magnitude are well within regulatory occupational exposure limits and do not represent an impact to worker health.

For members of the public, as discussed in Section 5.6.1 above, the licensee estimates that there will be no detectable increase in public dose during normal operations with the MOX LTAs. Use of the MOX LTAs in the reactor core will not change the characteristics of plant effluents or water use. During normal plant operation, the type of fuel material will have no effect on the chemistry parameters or radioactivity in the plant water systems. The fuel material is sealed inside fuel rods that are seal-welded and leaktight. Therefore, there would be no direct impact on plant radioactive effluents and the associated radiation exposure to members of the public.

3.5.3 Summary

Based on the NRC staff's review of the information provided in the licensee's application, the NRC staff concludes that there is reasonable assurance that the licensee will conduct its radiation protection and radioactive effluent release programs in a manner that maintains radiation exposures to plant workers and members of the public within the regulatory limits of 10 CFR 20.1301 and 10 CFR 20.1201.

3.6 Quality Assurance

3.6.1 Introduction

The licensee's application of February 27, 2003, included, in part, a description of the QA activities associated with the fabrication of the MOX LTAs by Framatome ANP, the supplier of the MOX LTAs to the licensee. This section of the SE addresses the programmatic aspects of the Framatome ANP QA program associated with the fabrication of the MOX LTA fuel pellets and fuel assemblies.

Section 3.5.4, "Quality Assurance," of the licensee's February 27, 2003, submittal contained a description of the QA process related to the fabrication and assembly of the MOX fuel pellets and fuel assemblies. As stated in the amendment request, Framatome ANP has the responsibility for the overall QA oversight of the entire fuel assembly fabrication process. As part of this effort, Framatome ANP will qualify every sub-vendor who operates under the technical requirements of the program and will verify that each sub-vendor and the sub-vendor's associated facilities meet the requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants". The applicant further stated that the qualification of these vendors and facilities shall include a combination of system audits conducted by Framatome ANP, review of audits performed by other Framatome ANP facilities, and surveillance audits by other approved Framatome ANP quality auditors.

3.6.2 Regulatory Basis

The NRC staff review of Section 3.5.4, "Quality Assurance," of the submittal was conducted in accordance with the review requirements described in Chapter 17, "Quality Assurance," of the SRP (Reference 68) to assure that the requirements of 10 CFR Part 50, Appendix B, were adequately implemented. The NRC staff used additional guidance provided in RG 1.28, Revision 3, 1985, "Quality Assurance Program Requirements (Design and Construction)," (Reference 69), ANSI/ASME Standard N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities," (Reference 71), and ANSI/ASME Standard NQA-1 1983, "Quality Assurance Requirements for Nuclear Facilities," (Reference 70) respectively, in its review.

The NRC staff customarily reviews and evaluates an applicant's description of its QA program for the design and construction phases in each application for a construction permit, a manufacturing license, or a standardized design certification in accordance with applicable portions of SRP 17.1. The acceptance criteria in this section are based on the relevant requirements of 10 CFR Part 50, Appendix B; 10 CFR Part 50, Appendix A; 10 CFR Part 50.55a; 10 CFR Part 50.55(e); and 10 CFR Part 50.34(a)(7) with emphasis on activities associated with the design and construction phases. The acceptance criteria deal with the QA controls related to the 18 areas outlined in 10 CFR Part 50, Appendix B, and review guidance embodied in the regulatory guidance referenced by SRP 17.1. Appendix B of 10 CFR Part 50 identifies all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. The 18 elements described in Appendix B specifically describe those planned and systematic actions.

3.6.3 Technical Evaluation

Since the fabrication of the LTAs is but one of the activities of a consortium effort that also includes development of a MOX FFF, it is considered useful to provide background information on the QA program for the MOX FFF. The review of the QA Program for the construction of the MOX FFF has been performed and documented in an NRC Evaluation Report dated October 1, 2001, (Reference 74). The scope of this SE is limited to the QA aspects associated with the fabrication of the MOX LTA's for use in the Catawba, Units 1 and 2 as described in the subject amendment application. Additionally, prior evaluations and approval of the Duke QA program and the Framatome Topical Report, "Framatome Quality Assurance Program (for United States Applications)," have been completed by the NRC staff and have been documented in letters from NRC to the applicants (References 75 and 76).

The NRC staff requested additional information from the licensee in letters dated August 13, 2003, and December 24, 2003, to support the current review. The focus of those requests was on the QA aspects of the MOX manufacturing process. Specifically, the NRC staff requested the following information pertaining to the scope of the Framatome ANP QA program: (1) a description of the Framatome ANP QA plans governing the fabrication activities affecting quality, (2) identification of individual sub-suppliers of materials to Framatome ANP and information pertaining to their QA programs and qualifications, and (3) information related to the various verification activities of Framatome ANP to ensure adequate implementation of the QA program for all fuel fabrication activities affecting quality. A discussion of these three areas follows:

(1) Description of Framatome ANP QA Plans

The licensee responded to the request for a detailed QA program description for the fabrication of the MOX LTAs, in its letter dated October 1, 2003. As part of its response, the licensee included a copy of the Framatome ANP manual, "Fuel Sector Quality Management Manual," (FQM Revision 1, US Version - Applicable July 2003), that defines the quality program that applies to the fabrication of components within Framatome ANP and items purchased from suppliers. The licensee provided supplemental information regarding the specific QA plan for the assembly and certification of the fuel rods and assemblies by letter dated February 2, 2004.

The FQM contains a detailed description of each of the Framatome ANP QA program attributes including criteria and requirements established to ensure compliance with the 18 criteria of a QA program described in 10 CFR Part 50, Appendix B. The NRC staff finds that the document is of sufficient detail to adequately identify specific actions, roles, and responsibilities within the Framatome ANP organization to assure that the scope and breadth of activities affecting quality are adequate. Additionally, the manual contains an evaluation of the Framatome ANP QA program attributes with respect to the NRC's SRP Section 17.1 and pertinent regulatory and

3.6.4 QA Summary

The NRC staff evaluated the scope of the QA activities involving the fabrication of the MOX LTA fuel pellets and fuel assemblies as described by the licensee, including the administrative controls governing those activities. The NRC staff finds that the proposed QA processes and activities described by the licensee in its amendment application as supplemented through letters dated October 1, 2003, and February 2, 2004, are consistent with the requirements of 10 CFR Part 50, Appendix B and the pertinent regulatory guidance described above and are, therefore, acceptable.

3.5 Security Plan

A non-safeguards information version of a safety evaluation will be provided in a supplement to this Safety Evaluation. The NRC staff's detailed conclusions will be provided in a document that, since it will contain safeguards information, will not be released to the public.

4.0 CONCLUSION

At the time of issuance of this SE, certain matters that are required to be completed to permit the issuance of any amendment to the operating licenses authorizing the use of MOX LTAs have not been completed. These include the completion of the NRC staff's review of the security plan as discussed in section 3.5 above, consultation with the State of South Carolina, and completion of the environmental consideration. The Commission has concluded, based on the considerations discussed in sections 1.0, 2.0 and 3.0 of this SE and subject to the completion of the matters discussed above, 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter, M. S. Tuckman, Duke, to NRC, "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", dated February 27, 2003.

2. Letter, R. E. Martin, NRC, to Duke, requesting additional information on McGuire spent fuel pool storage racks, dated July 14, 2003.

3. Letter, R. E. Martin, NRC, to Duke, requesting additional information on reactor systems, radiological consequences, and environmental impacts, dated July 25, 2003.

4. Letter, R. E. Martin, NRC, to Duke, requesting additional information on quality assurance, dated August 13, 2003.

5. Letter, M. S. Tuckman, Duke, to NRC, "Revision 16 to Duke Energy Corporation Physical Security Plan and Request for Exemption from Certain Regulatory Requirements in 10 CFR 11 and 73 to Support MOX Fuel Use," dated September 15, 2003.

6. Letter, M. S. Tuckman, Duke, to NRC, amending the February 27, 2003 application to apply only to Catawba Nuclear Station, dated September 23, 2003.

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