March 3, 2005

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

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB7863 AND MB7864)

Dear Mr. Barron:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 220 to Renewed Facility Operating License NPF-35 and Amendment No. 215 to Renewed Facility Operating License NPF-52 for the Catawba Nuclear Station (Catawba), Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 27, 2003, as supplemented. The amendments revise the TSs for Catawba to allow the use of four mixed oxide fuel lead test assemblies in one of the two Catawba units.

The application, as supplemented, also requested exemptions from certain NRC regulations. The document responding to those requests for exemptions is being issued from this office as separate correspondence.

A copy of the related Safety Evaluation is also enclosed. The enclosed Notice of Issuance will be published in the Federal Register.

Sincerely,

/RA/

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

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 220 to NPF-35
2. Amendment No. 215 to NPF-52
3. Safety Evaluation
4. Notice of Issuance

cc w/encls: See next page

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DUKE ENERGY CORPORATION

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 220
Renewed License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-35 filed by the Duke Energy Corporation, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees), dated February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3, November 4, December 10, 2003, and February 2 (two letters), March 1 (three letters), March 9 (two letters), March 16 (two letters), March 26, March 31, April 13, April 16, May 13, June 17, August 31, September 20, October 4, October 29 and December 10, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-35 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 220, which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 3, 2005

DUKE ENERGY CORPORATION

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 215
Renewed License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Corporation, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3, November 4, December 10, 2003, and February 2 (two letters), March 1 (three letters), March 9 (two letters), March 16 (two letters), March 26, March 31, April 13, April 16, May 13, June 17, August 31, September 20, October 4, October 29 and December 10, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 215, which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 3, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 220

RENEWED FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND LICENSE AMENDMENT NO. 215

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.7.16-1	3.7.16-1
3.7.16-4	3.7.16-4
4.0-1	4.0-1
4.0-2	4.0-2
5.6-5	5.6-5
B 3.7.16-1	B 3.7.16-1
B 3.7.16-2	B 3.7.16-2
B 3.7.16-3	B 3.7.16-3

SUPPLEMENT NO. 3 TO SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

AND THE OFFICE OF NUCLEAR SECURITY AND INCIDENT RESPONSE

RELATED TO AMENDMENT NO. 220 TO RENEWED FACILITY

OPERATING LICENSE NPF-35 AND

AMENDMENT NO. 215 TO 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

1.1 License Amendment Request

By letter to the Nuclear Regulatory Commission (NRC) dated February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3, November 4, December 10, 2003, and February 2 (two letters), March 1 (three letters), March 9 (two letters), March 16 (two letters), March 26, March 31, April 13, April 16, May 13, June 17, August 31, September 20, October 4, October 29 and December 10, 2004, Duke Energy Corporation, et al. (Duke, the licensee), submitted a request for changes to the Catawba Nuclear Station (Catawba), Units 1 and 2, Technical Specifications (TSs). The requested changes would revise TS 3.7.16, “Spent Fuel Assembly Storage;” TS 4.2, “Reactor Core;” TS 4.3, “Fuel Storage;” and TS 5.6.5, “Core Operating Limits Report [COLR],” to allow the use of four mixed oxide (MOX) fuel lead test assemblies (LTAs) in one of the two Catawba units.

1.2 Atomic Safety and Licensing Board Hearing

1.2.1 Safety and Environmental Related Issues

On July 25, 2003, a Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing was published in the *Federal Register* (FR) (68 FR 44107). Following that notice, the Nuclear Information and Resources Service (NIRS) and the Blue Ridge Environmental Defense League (BREDL) submitted, on August 21 and 25, 2003, respectively, petitions to intervene and requests for hearing pursuant to 10 CFR 2.714.

On September 17, 2003, an Atomic Safety and Licensing Board (hereinafter, the Board) (ASLBP No. 03-815-03-OLA) was established to preside over the proceeding (68 FR 55414; September 25, 2003).

In an Order issued on March 5, 2004 (ASLBP No. 03-815-03-OLA, LBP-04-04), the Board reframed and admitted three BREDL contentions (I, II, and III) and admitted BREDL as a party to the proceeding. All of NIRS's contentions were rejected and NIRS was not admitted as a party to the proceeding. On May 6, 2004, the Board dismissed Contention III (See LBP-04-07). On May 25, 2004, the Board granted BREDL's request to withdraw Contention II. On July 14 and 15, 2004, the Board held a hearing on the single admitted safety-related contention. This contention was related to the adequacy of the loss-of-coolant accident (LOCA) analysis performed to support the use of the MOX LTAs. On December 22, 2004, the Board issued a Partial Initial Decision with respect to this matter finding that there is reasonable assurance that operation of Catawba with the four MOX LTAs will not endanger the health and safety of the public.

1.2.2 Physical Security Plan Related Issues

In addition to the non-security related contentions addressed above, various security related issues have arisen. On September 15, 2003, Duke submitted proposed revisions to its physical security plan (PSP) related to the use of MOX LTAs. The NRC staff issued a supplement to its MOX Safety Evaluation (SE) addressing the proposed PSP revisions on May 5, 2004. BREDL submitted its security related contentions on March 3, 2004.

The Board held a hearing on the single admitted security contention during the week of January 11-14, 2005. This contention is related to the adequacy of the provisions undertaken by Duke to provide protection of the MOX LTAs. Findings and reply findings of fact and conclusions of law were filed on February 1 and February 8, 2005, respectively. An ASLB decision on the security contention was pending when this supplemental SE was issued.

2.0 SAFETY EVALUATION

This SE, Supplement No. 3, was preceded by an original SE and two other supplements to the original SE. The original SE issued on April 5, 2004 (Reference 2), addressed the areas of reactor systems, radiological dose consequences, spent fuel pool (SFP) cooling, reactor vessel materials, occupational dose and routing effluents, and quality assurance. The reactor systems area included a description of the MOX LTA mechanical design features, the effects of MOX LTAs on plant operations nuclear design, thermal-hydraulic and mechanical design, the impact of MOX LTA on LOCA analyses and non-LOCA analyses, a criticality evaluation of MOX fuel storage in the SFP, and an evaluation of the specific changes to the TSs. The dose consequence analysis included a discussion of the consequence analyses, the MOX LTA fission product inventory, the impact of MOX on fuel rod gap fractions, the potential core damage accidents during power operations, and fuel-handling accidents (FHA). Based on the considerations discussed in the original SE and subject to the completion of certain matters that were incomplete at that time, the NRC staff concluded 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.

The physical security matters were incomplete at the time the original SE was issued and those matters were addressed in Supplement No. 1 to the SE that was issued on May 5, 2004. Supplement No. 1 provided the NRC staff's evaluation of the changes to the PSP for the protection of MOX fuel. It also provided the NRC staff's evaluation of Duke's request for exemptions from certain parts of NRC regulations related to the PSP for Catawba. Based on its review, the NRC staff found that additional protective measures proposed by Duke will provide enhanced physical security for MOX LTAs, beyond those measures that are currently in place for low-enriched uranium (LEU) fuel such that there is a reasonable asurance that public health and safety and security will be maintained. The NRC staff also approved Duke's proposed revision to the PSP for Catawba, dated April 13, 2004. On October 29, 2004, the NRC staff issued a letter concerning the licensee's compliance with the NRC staff's revised design basis threat Order (DBT Order). The DBT Order was issued on April 29, 2003, and was concerned with overall protection of the Catawba plant and was not specifically related to the licensee's application for the use of MOX LTAs. The NRC staff's letter of October 29, 2004, noted that the provisions for the protection of MOX were included in Appendix E of the PSP that had been updated to meet the requirements of the DBT Order. The NRC staff has examined Appendix E and finds that it contains the provisions for the protection of MOX LTAs that the NRC staff approved in its SE supplement issued on May 5, 2004. Accordingly, the NRC staff approves Appendix E to the PSP for the protection of the MOX LTAs, as referenced in the NRC staff's letter dated October 29, 2004.

Another matter, related to the Westinghouse Next Generation Fuel (NGF) design, that was identified subsequent to the original SE issuance, was discussed in Supplement No. 2 to the SE that was issued on July 27, 2004 (Reference 4). It provided the NRC staff's evaluation of Duke's use of eight LTAs of the Westinghouse NGF design in parallel with the use of the four MOX LTAs. Based on its review of the Duke and Westinghouse NGF LTA analyses, the NRC staff found that the effect of the eight NGF LTAs on the four MOX LTAs was conservatively evaluated and the NGF assemblies will not have any effect on the MOX LTAs.

The NRC staff's original SE (Reference 2), Supplement No. 1 to the SE (Reference 3), and Supplement No. 2 to the SE (Reference 4), are incorporated by reference into this SE.

2.1 Radiological Dose Consequence Analysis

A further matter identified subsequent to the issuance of the original SE was identified by Duke. In its letter dated August 31, 2004 (Reference 7), Duke informed the NRC staff that certain radiological dose consequence information used to support the use of 4 MOX LTAs, had been based on Updated Final Safety Analysis Report (UFSAR) information that was out of date[1]. Duke's letter dated September 20, 2004 (Reference 8), provided technical analysis information addressing this issue. The NRC staff requested additional information (RAI) in its letter dated October 7, 2004 (Reference 9), and Duke responded in its letter dated October 29, 2004 (Reference 11). Duke provided further information on this matter in its letter dated December 10, 2004 (Reference 12).

[1] The NRC staff considered enforcement action related to this issue and the issue discussed above related to NGF fuel, and published its results in a Notice of Violation dated January 24, 2005.

2.1.1 Regulatory Evaluation

The basis for the NRC staff's regulatory evaluation is unchanged from that presented in Section 3.1, "Regulatory Evaluation," of the NRC staff's original SE (Reference 2).

2.1.2 Technical Evaluation

2.1.2.1 Radiological Dose Consequences

As discussed in the NRC staff's original SE (Reference 2), Duke considered the impact of the four MOX LTAs on each design basis accident (DBA) previously analyzed in the Catawba UFSAR. Duke's approach to evaluating the potential impact of the MOX LTAs was to compare the relative differences in radionuclide inventory and determine adjustment factors that account for these differences. These adjustment factors were applied to the UFSAR analysis results for these events. These factors consider the increase in the fission product inventory of radioiodine in MOX fuel, the increased transfer of fission products to the fuel gap region, and the fraction of the number of damaged fuel assemblies that are MOX LTAs. The magnitude of the adjustment factor corresponds to the relative impact of the four MOX LTAs on the previous DBA analysis results. The NRC staff reviewed the adjustment factors developed by Duke, performed independent calculations, and concluded that the factors were proper and acceptable. The NRC staff reviewed the application of the factors to the previous DBA analysis results and confirmed Duke's conclusion that the radiological consequences of the DBAs would continue to meet regulatory requirements with the four MOX LTAs. That NRC staff review of the development and application of these adjustment factors is reported in Section 3 of the original SE (Reference 2).

By letter dated August 31, 2004, Duke notified the NRC staff that it had determined that information in Table 15-[14] of the Catawba UFSAR was outdated. Since Duke had used this information in preparing its February 27, 2003, submittal, this meant that some of the radiological dose consequence analysis information provided in that submittal and in certain supplemental letters, was in error. Although the relative magnitude of the MOX LTA impact was not affected, the absolute values of the final doses for the affected DBAs were in error. Duke submitted corrected information in a letter dated September 20, 2004, and supplemented this information by letters dated October 29 and December 10, 2004. These later submittals indicated that the subject outdated information applied to the the DBA analysis results for the rod ejection accident (REA) and the locked rotor accident (LRA).

In addition to the revised REA and LRA analyses, Duke updated the LOCA control room dose analysis. The fuel handling accident (FHA), the weir gate drop (WGD) accident, and the fresh MOX LTA drop accident analyses discussed in the original SE (Reference 2) were not affected.

Section 3.2.5 of the NRC staff's original SE is the only section of that SE that is affected by the issues addressed in Duke's letters dated August 31, September 20, October 29, and December 10, 2004. Table 3 of this supplemental SE discusses the matters in the text of the original SE, Section 3.2.5 that were affected by these changes. This supplement to the SE discusses the NRC staff's review of the changes incorporated into the revised REA, LRA, and the LOCA dose analyses. The NRC staff performed independent calculations to confirm the results of the radiological dose analyses for the REA, LRA, and LOCA control room doses. In the following discussion, "original analysis" refers to the analysis described in the UFSAR, while "current analysis" refers to Duke's current analysis of record. Duke did scale some of the results to reflect data developed subsequent to the current analysis of record, for example, results of control room in leakage testing performed in 2003.

Rod Ejection Accident

The accident considered is the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding is projected due to the reactivity spike. Analysis assumptions include a reactor trip, actuation of safety injection, and a loss of offsite power (LOOP) concurrent with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. The release to the environment is assumed to occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage) and

- Release of reactor coolant system (RCS) inventory via primary-to-secondary leakage through steam generators.

While the actual doses from a REA would be a composite of the two pathways, an acceptable dose from each pathway, modeled as if it were the only pathway, would show that the composite dose would also be acceptable.

This event was analyzed as described in the UFSAR. The analysis was recalculated in support of subsequent plant modifications that replaced steam generators at Catawba, Unit 1 and modified the auxiliary feedwater system at both units. The UFSAR should have been updated to reflect the recalculated analyses but was not. As a consequence, Duke used outdated thyroid dose data in its assessment of the impact of inserting the four MOX LTAs in the core. In letters dated September 20, October 29, and December 10, 2004 (References 8, 11, and 12), Duke tabulated the current analysis assumptions and inputs and provided an explanation of the differences from the UFSAR analysis. As a result of including these modifications and other changes reported in References 8, 11, and 12, the reported thyroid dose at the exclusion area boundary (EAB) for LEU fuel increased from 1.0 rem for each unit to 21.8 rem for Unit 1 and 30.7 rem for Unit 2. This is shown in the attached Table 4, which is based on Table 4 from the original SE (Reference 2) and on Duke's Attachment 2 in Reference 12. As also shown in Table 4 of this supplemental SE, the inclusion of the four MOX LTAs increases the EAB thyroid dose from 21.8 to 22.3 rem for Unit 1 and from 30.7 to 31.5 rem for Unit 2.

The secondary release pathway models the transport of the activity released from the damaged fuel to the environment via leakage from the RCS to the secondary plant via steam generator tube leakage. NRC staff analysis guidance allows licensees to assume some retention of the released iodine within the steam generator bulk water, provided that the tubes are completely submerged. Duke's updated accident transient analysis indicated that the tubes would be uncovered for a period of time following the accident. As such, the current radiological analyses could not credit iodine mitigation during this period, resulting in significant increases in the postulated releases and doses. Since the original analyses did not postulate tube uncovery, this analysis change accounts for the major portion of the increase in accident doses. Duke's transient analysis was performed using analysis methodologies previously approved by the

NRC staff. As such, the NRC staff finds Duke's estimate of the tube uncovery periods to be acceptable. Other analysis changes are addressed below.

1) The original analysis was performed conservatively assuming a reactor power of 3565 mega-watts thermal (MWt); the current analysis was performed assuming 3479 MWt. Since the current value represents the licensed reactor power plus a two percent uncertainty, this change is acceptable.

2) The original analysis source term included the initial RCS specific activity at the onset of the accident; the current analysis did not. Duke stated that since the dose consequences of an REA are dominated by the fission product release from the fuel, this negligible contributor to the dose could be omitted. Based on its experience in other reviews, the NRC staff concurs and finds this change acceptable.

3) The original analysis assumed that 30 percent of the core inventory of Krypton-85 was in the fuel rod gap along with 10 percent of the core inventory of other fission products. The current analysis assumes a gap fraction of 10 percent of all noble gases and halogens. The revised assumption is consistent with the guidance in Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," and is, therefore, acceptable.

4) The original analysis assumed a constant value of steam generator tube leakage of 360 gallons per day (gpd) per steam generator (a total of 1 gpm as provided by the original technical specifications); the current analysis assumes an initial steam generator tube leakage of 150 gpd per steam generator based on the current technical specification allowable leakage. In addition, the current analysis assumed that the leakage rate varies by time based on the accident transient analysis. Duke's transient analysis was performed using analysis methodologies previously approved by the NRC staff. The NRC staff finds this change acceptable.

5) The original analysis used EAB and low population zone (LPZ) atmospheric dispersion factor (X/Q) values that were subsequently recalculated (prior to the current analysis). The current analysis used updated values. The X/Q values are discussed later in this SE.

6) Duke's original analysis did not assess the control room doses due to an REA; the current analysis does assess the control room dose. The current analysis assumed a control room in-leakage of 10 cubic feet/minute (cfm). Duke scaled the current analysis results for the present effort to reflect the recent tracer gas test result of 30 cfm. The staff questioned Duke on the different values of control room pressurization and recirculation filter flow rates used in the REA analysis and in other DBA analyses. Duke provided additional information that supports a determination that the individual DBA analyses remain conservative. The NRC staff finds the control room analysis acceptable. The control room X/Q values are discussed later in this SE.

7) The original analysis assumed that 10 percent of the core exceeded departure from nucleate boiling (DNB) and that 0.25 percent of the core melted as a result of the REA. The current analysis assumed that 50 percent of the core exceeded DNB and that no melting occurred. The current assumptions were previously found acceptable by the

NRC staff in its November 15, 1991, approval of Duke topical report DPC-3001 (Reference 13).

8) The original analysis did not assess the radiological consequences of leakage from emergency core cooling systems (ECCS) outside of containment; the current analysis does. Since the NRC staff analysis guidance in RG 1.77 and Standard Review Plan (SRP) Section 15.4.8.A does not require analysis of this dose contributor, the NRC staff finds Duke's inclusion of this pathway conservative and, therefore, acceptable.

9) The original analysis assumed a containment volume that did not include the volume of the ice condenser; the volume used in the current analysis does. This parameter has a negligible impact on the dose analysis since the modeling of the fission product transport to the environment is independent of the containment volume (i.e., release expressed in terms of percent per day). The NRC staff finds this change to be acceptable.

10) The original analysis assumed an annulus ventilation system flow rate of 9000 cfm that was reduced by 15 percent at 15 minutes; the current analysis assumes a flow rate of 8100 cfm. The 8100 cfm value is the lower bound of the surveillance test acceptance criterion. The flow reduction was linked to assumed use of the containment hydrogen purge system. This system would only be used if both trains of equipment provided to reduce hydrogen in containment were to fail. Since assuming two failures is beyond the design basis, the NRC staff finds that the flow reduction assumption is conservative. The NRC staff finds these changes to be acceptable.

11) The original analysis assumed operation of the containment spray system as a fission product removal mechanism; the current analysis does not credit this mitigation. Since not crediting sprays is conservative, the NRC staff finds this assumption to be acceptable.

12) The original analysis assumed only a two minute release via the steam dump; the current analysis assumes that the plant is cooled down to decay heat removal system conditions within 330 minutes. Following this, the steam release is terminated as further cooldown is completed by the decay heat removal system. The steam release is obtained by a heat balance that accounts for initial stored energy, decay heat removal, and cooldown. This change increased the release duration and, thus, resulted in increased doses. The extended release is consistent with NRC staff analysis guidance and is acceptable.

13) A radial peaking factor is used in radiological analyses to adjust the fission product inventory to reflect the difference between fuel assembly reactor power and core-average reactor power. This correction is significant when the amount of damaged fuel is a small fraction of the entire core. Duke's REA analysis assumes a radial peaking factor of 1.0 since, given the projected 50 percent core damage, the fission product inventory corresponding to the core-average power is appropriate and no correction is warranted. The NRC staff finds this assumption to be acceptable.

<u>Locked Rotor Accident</u>

The accident considered is the instantaneous seizure of a reactor coolant pump rotor (i.e., a locked rotor accident) that causes a rapid reduction in the flow through the affected RCS loop. Analysis assumptions include a reactor trip, actuation of safety injection, and a LOOP concurrent with the reactor trip. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. As the LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment.

This event was analyzed as described in the UFSAR. As with the REA addressed above, the analysis was recalculated in support of subsequent plant modifications that replaced steam generators at Catawba, Unit 1 and modified the auxiliary feedwater system at both units. The UFSAR should have been updated to reflect the recalculated analyses but was not. As a consequence, Duke used outdated thyroid dose data in their assessment of the impact of inserting the four MOX LTAs in the core. In letters dated September 20, October 29, and December 10, 2004 (References 8, 11, and 12), Duke tabulated the current analysis assumptions and inputs and provided an explanation of the differences from the UFSAR analysis. As a result of including these modifications and other changes reported in References 8, 11 and 12, the reported thyroid dose at the EAB for LEU fuel increased from 3.7 rem for each unit to 23.6 rem for Unit 1 and 22 rem for Unit 2. This is shown in the attached Table 4, which is based on Table 4 from the original SE and on Duke's Attachment 2 in Reference 12. As also shown in Table 4 of this supplemental SE, the inclusion of the four MOX LTAs increases the EAB thyroid dose from 23.6 to 26.9 rem for Unit 1 and from 22.0 to 27.8 rem for Unit 2.

The secondary release pathway models the transport of the activity released from the damaged fuel to the environment via leakage from the RCS to the secondary plant via steam generator tube leakage. NRC staff analysis guidance allows licensees to assume some retention of the released iodine within the steam generator bulk water, provided that the tubes are completely submerged. Duke's updated accident transient analysis indicated that the tubes would be uncovered for a period of time following the accident. As such, the current radiological analyses could not credit iodine mitigation during this period, resulting in significant increases in the postulated releases and doses. Since the original analyses did not postulate tube uncovery, this analysis change accounts for the major portion of the increase in accident doses. Duke's transient analysis was performed using analysis methodologies previously approved by the NRC staff. As such, the NRC staff finds Duke's estimate of the tube uncovery periods to be acceptable.

The analysis changes discussed in Items 1 through 6 above for the REA, are also applicable to the LRA analysis. In addition, the original LRA analysis assumed that 10 percent of the core exceeded DNB as a result of the REA; the current analysis assumed that 9.5 percent of the core exceeded DNB for a LRA at Unit 1 and 5.0 percent at Unit 2. The amount of fuel that is projected to fail can vary from cycle to cycle and is determined using transient analysis methods that were approved by the NRC staff in previous licensing proceedings. These core damage projections are expected to be conservative in that previous and current cycle-specific assessments resulted in no calculated core damage. Duke committed to perform cycle-specific checks on the calculated core power peaking to ensure that the projected core damage is bounded by the core damage assumptions used in the dose analyses. The NRC staff finds these assumptions and commitments to be acceptable.

Control Room Doses due to LOCA

The accident considered is a double-ended rupture of the largest pipe in the RCS. The objective of this postulated DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that the ECCS is not effective in preventing core damage. A LOCA is a failure of the RCS that results in the loss of reactor coolant that, if not mitigated, could result in fuel damage including a core melt. The primary coolant will blow down through the break to the containment, depressurizing the RCS and pressurizing the containment. A reactor trip occurs and the ECCS is actuated to force borated water into the reactor vessel. Containment sprays actuate to depressurize the containment. Thermodynamic analyses, done using a spectrum of RCS break sizes, show that the ECCS and other plant safety features are effective in preventing significant fuel damage. Nonetheless, the radiological consequence portion of the LOCA analysis conservatively assumes that ECCS is not effective and that substantial fuel damage occurs. For these analyses, the failure of the largest pipe in the RCS is postulated since this represents the larger challenge to mitigating the radionuclide releases.

The UFSAR discussion of the DBA LOCA doses at the EAB and LPZ corresponded to the current analysis of record. In the following discussion, two types of leakage are mentioned including (a) leakage from ECCS components outside of containment and (b) an assumption of a gross failure of a passive component of the ECCS recirculation piping as addressed by SRP Section 15.6.5, Appendix B. The current analysis for the control room dose did not assess the radiological consequences of leakage from ECCS outside of containment; the EAB and LPZ analysis did consider this release pathway. Duke conservatively adjusted the control room dose results to reflect the dose associated with this release pathway. In performing this adjustment for ECCS leakage, Duke used the design basis assumptions used in the assessment of the offsite doses with one exception regarding the magnitude of the leakage rate from the ECCS system. NRC staff guidance requires an applicant to assume a leakage rate equal to twice the expected design leakage starting with the onset of recirculation flow and continuing for 30 days. If the plant design does not include an engineered safeguards feature (ESF) atmospheric filtration system, the applicant is also required to assume a gross failure of a passive component of the ECCS recirculation piping. The auxiliary building filtered ventilation exhaust system (ABFVES) was licensed as an ESF atmosphere filtration system. In its SE related to the operation of the Catawba units (NUREG-0954), the NRC staff concluded that the ABFVES met the acceptance criteria of Standard Review Plan 9.4.3. As such, the original licensing offsite dose analyses did not model a gross failure of passive components of the ECCS recirculation piping. However, this analysis conservatively did not credit any radionuclide filtration performed by the ABFVES to reduce the environmental release from the assumed design leakage.

Subsequent to the original licensing, Duke re-analyzed the offsite doses due to ECCS leakage and assumed a gross, passive failure. As before, no credit was taken for radionuclide filtration by the ABFVES. This analysis was described in an update to the UFSAR and became the current analysis of record. Although no credit was taken for the ABFVES with regard to dose analyses, the ABFVES design basis configuration was never revised. The ABFVES continues to be treated as an ESF atmosphere filtration system and is subject to TS operability and ventilation filter test program requirements. This was confirmed by Duke in its letter dated December 10, 2004.

For the re-assessment of the control room dose, Duke returned to the original licensing basis and did not model a gross, passive failure.[2] As before, no credit was taken for radionuclide filtration by the ABFVES. Other assumptions from the offsite analysis were used unchanged. Since these ECCS leakage assumptions were previously found to be acceptable, the NRC staff finds the application of these assumptions to control room doses (including the assumption regarding a gross passive failure) to be acceptable, provided the ABFVES continues to be treated as an ESF atmospheric filtration system subject to the current TS operability and ventilation filter test program requirements.

The current analysis assumed a control room in-leakage of 10 cfm. Duke scaled the current analysis results for the present effort to reflect the recent control room tracer gas test result of 30 cfm. The NRC staff questioned Duke on the different values of control room pressurization and recirculation filter flow rates used in the LOCA analysis and in other DBA analyses. Duke provided additional information that supports a determination that the individual DBA analyses remain conservative.

As a result of the changes discussed above, Duke determined that the resulting control room thyroid dose after a postulated LOCA, considering the use of LEU fuel would be 12.8 rem, and that considering the use of four MOX LTAs increases this dose to 13.0 rem. The NRC staff finds the updated control room analysis to be acceptable since these values are below the acceptance criterion as shown in Table 4.

Atmospheric Dispersion Factors

The licensee used previously calculated atmospheric dispersion factors (χ/Q values) in the Catawba MOX dose assessment. These values, provided in Reference 11, are listed in Table 1.

The 0–2 hour EAB and 0–8 hour LPZ X/Q values are the licensing basis values that were previously approved as part of Amendments 159 and 151 dated April 29, 1997. The 8–24 hour, 1–4 day, and 4–30 day LPZ X/Q values were previously calculated by the licensee utilizing the same general methodology used to generate the shorter time period X/Q values that were approved by the NRC staff in Amendments 159 and 151. The NRC staff performed an independent assessment of the licensee's estimates using the PAVAN computer code with the hourly onsite meteorological data discussed in the SE associated with Amendments 198 and 191 dated April 23, 2002. The PAVAN computer code implements the guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." Based upon the results of these calculations that were bounded by the licensee's estimates, an examination of other prior NRC staff estimates, and review of the information provided by the licensee, the NRC staff found the χ/Q values calculated by the licensee acceptable.

The control room X/Q values are the licensing basis values presented in the Catawba UFSAR that were calculated using the Murphy-Campe methodology to support initial licensing of the Catawba facilities. As discussed in the October 29, 2004, response to an RAI, postulated

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[2] This change in assumptions also resulted in a change in the 0-30 day LPZ dose as shown in Table 4. This change did not impact the 0-2 hour EAB dose.

releases from the LOCA, REA, and LRA were all assumed to occur from the same bounding location to the environment. Since the design basis assessment assumes a single limiting failure, the 0–10 hour X/Q values used in the LOCA dose assessment were increased by a factor of two to account for the possible failure of one of the dual control room air intakes. The inflow rate of both intakes was assumed to be equal, with the intake drawing the uncontaminated air failing at the beginning of the release and the failure being remedied within 10 hours. For the postulated LRA and REA, both control room air intakes were assumed to function during the entire accident since the limiting failure, the loss of an auxiliary feedwater pump, would result in a higher dose than the failure of an intake. The licensee stated that there is no common failure mode that would simultaneously incapacitate both an auxiliary feedwater pump and a control room air intake. Based upon the information provided by the licensee, the NRC staff found the X/Q values in Table 1 to be acceptable for use in its MOX dose assessment.

2.1.2.2 Reactor Systems Analysis

Duke's letter dated September 20, 2004, indicated that certain baseline LEU doses were based on outdated information from the UFSAR and that the problem affects only the offsite and control room doses for the REA, LRA, and the LOCA. In order to confirm that this outdated information in the UFSAR does not affect the results of the thermal-hydraulic licensing analyses of the REA and LRA that were used to provide the inputs for the dose calculation for these events, the NRC staff requested (Reference 9) that the licensee provide additional information that would: 1) identify any outdated information in the current UFSAR for Section 15.3.3 (LRA) and Section 15.4.8 (REA) and discuss their effects on the results of the analyses for these events, and 2) provide detailed results of the cycle specific thermal-hydraulic analyses for the REA and LRA considering the calculated number of fuel pins that experience DNB and steam generator water levels during these transients.

In response to the staff RAIs, Duke's letter dated October 29, 2004, identified three outdated items in the UFSAR involving the LRA and REA. They are: 1) in Section 15.3.3.2, the description of modeling of the pressurizer code safety valves is outdated. This modeling was revised in Revision 2 to the Duke topical report DPC-NE-3002-A (Reference 14) and the current cycle specific analysis is consistent with the methodology documented in this NRC-approved topical report; 2) on page 15.4-26, the Doppler and moderator temperature coefficient adjustment reference should be UFSAR Reference 25 instead of Reference 15, which is an outdated methodology; and 3) on page 15.4-31, UFSAR Reference 25 should be Revision 2, dated December 18, 2002. However, this outdated information in the UFSAR does not affect the results of the current cycle specific thermal-hydraulic analyses. The licensee has committed to update this information in the UFSAR, as required by 10 CFR 50.71(e).

The licensee's thermal-hydraulic analyses of the LRA and REA were performed using the NRC staff-approved methodology and computer code (RETRAN-02). The analyses included a bounding analysis to determine the duration of steam generator tube uncovery, and a cycle specific analysis of the core power distribution to determine the number of fuel pins experiencing DNB. Those fuel pins experiencing DNB are assumed to fail. Due to the difference in steam generator design, a separate thermal-hydraulic analysis was performed for each of the two Catawba units. Also, a different approach to quantify the tube bundle uncovery time is used for the two units. The method used for Unit 2 is based on the tube covered with liquid water and the other unit is based on the tube covered with the mixture of water and voids. The NRC staff considers both approaches acceptable.

In its response to Reactor Systems RAI No.1(b) (Reference 11), the licensee provided the results of its current cycle specific analyses including the duration of the tube bundle uncovery and the calculated number of failed fuel pins for LRA and REA. As discussed above, the data regarding the duration of the tube bundle uncovery is considered in the NRC staff's assessment of radiological consequences as all fission products entrained with primary to secondary leakage are released to the environment while the tube bundles are uncovered. The results of the licensee's analyses show that the calculated amount of failed fuel in a LRA is 0 percent for both units for two recently analyzed fuel cycles. Compared to the bounding value of 9.5 percent for Unit 1 and 5.0 percent for Unit 2 in the radiological assessment for the LRA, the operation of the current cycle is bounded by the radiological assessment of record for the LRA. Also, the results of the licensee's analyses show that the calculated fuel failure is 14.6 percent for Unit 1 and 15.7 percent for Unit 2 in REA for two recently analyzed fuel cycles. Compared to the bundling value of 50 percent failed fuel assumed for both units in the REA analysis, the operation of the current cycle is bounded by the radiological assessment of record for REA.

The licensee stated, in its letter dated October 29, 2004, that if future cycle specific reload analyses do not meet the associated bounding limits on fuel failure as established above, the licensee will modify its analysis and/or adjust the design so that the analyzed results are within the established limits. The NRC staff finds that to be acceptable.

2.1.2.3 Conclusion for Outdated Radiological Dose Consequence Information

The NRC staff has determined that the licensee's updates of the offsite doses for the LRA, REA, and the LOCA control room doses were performed with acceptable analysis assumptions and methods assuming LEU reactor fuel. Analysis parameters found acceptable to the NRC staff for these analyses are tabulated in Table 2. Table 3 compares the dose from the licensee's initial analyses to the updated analyses and reflects a comparison of the licensee's results to those of the NRC staff's independent analysis. In the Table 4 "All LEU Core" column, the values in parentheses from the original SE are compared to current LEU values provided by the licensee. Comparison of the Table 4 "All LEU Core" and "LEU Core Plus 4-MOX LTAs" shows the relative effect of the 4 MOX LTAs.

In 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 identified and dispositioned above, the NRC staff finds that Duke used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.1 of the original SE. The NRC staff compared the doses estimated by Duke to the applicable criteria identified in Section 2.1 of the original SE. These doses are summarized in Table 4 of this report. Based on its review, as documented above, the NRC staff finds that the EAB, LPZ, and control room doses from postulated DBAs will continue to meet the acceptance criteria identified in Section 2.1 of the original SE. Therefore, use of four MOX LTAs at either Catawba unit is acceptable with regard to the radiological consequences of postulated DBAs.

2.2 TECHNICAL SPECIFICATION CHANGES

The changes to TS 3.7.16, “Spent Fuel Assembly Storage,” TS 4.2, “Reactor Core,” TS 4.3, “Fuel Storage” and to TS 5.6.5, “Core Operating Limits Report,” were evaluated in Section 2.6, “Technical Specification Changes,” of the original SE, and were found to be acceptable. On this basis, and other considerations, as discussed in this supplement to the original SE, the NRC staff concludes that the proposed changes to the Catawba TSs to permit the use of four MOX LTAs in one of the Catawba units are acceptable.

3.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission’s regulations at 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration (NSHC) if the operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of NSHC in its February 27, 2003, amendment request. The NRC staff reviewed the licensee’s analysis and, based on its review that also considered additional analysis performed by the NRC staff, it appeared that the three standards of 10 CFR 50.92(c) were satisfied. Therefore, the NRC staff proposed to determine that the amendment request involved a NSHC, and published its proposed determination in the *Federal Register* for public comment on July 12, 2004 (69 FR 41852).

The NRC staff has determined that the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a significant reduction in a margin of safety. The following NRC staff evaluation, in relation to the three standards of 10 CFR 50.92 (c) supports the NRC staff’s final NSHC determination.

I. Probability and Consequences Evaluation

The proposed license amendment to allow the use of 4 MOX LTAs does not involve a significant increase in the probability or consequences of an accident previously evaluated. The “accidents” previously evaluated are described in the UFSAR and fall into one of the following four categories:

- Normal Operation and Operational Transients
- Faults of Moderate Frequency
- Infrequent Faults
- Limiting Faults

Inspection of the UFSAR descriptions reveals that the presence of MOX LTAs could potentially impact the probability of occurrence for only two “accidents;” Radioactivity in Reactor Coolant Due to Cladding Defects and FHA Accidents. An evaluation of each of these events follows.

Radioactivity in Reactor Coolant Due to Cladding Defects Probability

Cladding defects are imperfections in the cladding material of a fuel assembly that allow fission products from the active fuel material to migrate to the reactor coolant. They can be caused by manufacturing defects that go undetected until the stresses of pressure, temperature, and/or irradiation eventually result in fuel cladding failure. This type of cladding failure occurs very infrequently in low-enriched uranium (LEU) fuel. The Mark BW design, which is the basis for the Mark BW/MOX1 design to be used in the MOX LTAs, has experienced a failure rate of less than one per 100,000 rods, from all manufacturing related causes, since its inception in 1987. There is no reason to expect that the probability of this type of failure in a MOX fuel assembly will be any different than for a LEU fuel assembly because the probability of fuel failure due to these factors is no different for MOX fuel assemblies than for LEU fuel assemblies. The MOX LTAs will be manufactured using the same quality standards that are used in the manufacture of LEU fuel, under a Quality Assurance program that conforms to 10 CFR 50, Appendix B. Likewise, the same operational procedures and precautions to preclude loose parts and debris in the reactor coolant will equally preclude fuel failures from these mechanisms for the MOX and LEU fuel assemblies.

Other mechanisms that could potentially cause fuel cladding failure are physical interaction of the cladding with loose debris in the reactor coolant system or corrosion product transport and buildup on cladding material. The design of both the current LEU fuel assemblies and the planned MOX fuel assemblies minimizes these types of interactions such that the probability of fuel failure is equally unlikely for both MOX and LEU fuel assemblies.

<u>Fuel Handling Accident Probability</u>

There is nothing in the physical design of a MOX fuel lead assembly that would make it more susceptible to a fuel handling accident than a LEU assembly. The physical dimensions are virtually identical, the difference in weight between a MOX assembly and an LEU assembly is less than 1 percent and the top nozzle engages the manipulator crane and handling fixture in the same manner as LEU fuel.

The shipping container and associated unloading procedure for a fresh MOX LTA are slightly different from that of a LEU fuel assembly but such differences do not result in a significant increase in the probability of an accident. The MOX LTA shipping container is an end-loaded container with capacity for one fuel assembly as opposed to a LEU shipping container that is side loaded and has the capacity for two fuel assemblies. The MOX LTA container is unloaded by uprighting the container, removing the closure lid, grappling the assembly with the Fuel Handling Tool, and lifting the assembly with a straight vertical lift out of the container. This is a straightforward lifting operation that will be practiced in a dry run involving a dummy fuel assembly, the MOX fuel shipping package, and specific fuel handling procedures. The same plant equipment will be used to grapple and lift a MOX fuel assembly that is used to lift a LEU fuel assembly. Once the MOX LTAs are unloaded and placed into the spent fuel pool, subsequent handling operations are identical to LEU fuel handling. Thus, it is concluded that the probability of a fuel handling accident involving a MOX fuel assembly drop, either inside containment or inside the fuel building, is no different than for a LEU assembly.

The other scenarios considered as part of the fuel handling accident analyses are a weir gate drop into the spent fuel pool and a tornado-generated missile entering the spent fuel pool. There is no connection between the type of fuel assembly and the probability of occurrence of either of these accidents. The probability of a tornado missile entering the spent fuel pool is a natural event whose frequency of occurrence will not change with the storage of MOX fuel assemblies in the fuel pool. The probability of dropping a weir gate into the spent fuel pool is dependent on the reliability of handling fixtures, crane rigging procedures, and the number of handling operations, none of which will be affected adversely by the handling or presence of MOX fuel assemblies.

The conclusion is that amending the Catawba licenses to allow the receipt, handling, storage, and use of MOX fuel lead assemblies does not result in a significant increase in the probability of occurrence of any accident previously evaluated in the UFSAR.

NRC Staff Analysis of Consequences

The licensee's calculated numerical values of dose consequences have changed since the licensee's initial submittal as addressed in the licensee's submittals dated November 3, 2003, March 1, March 16, August 31, September 20, October 29 and December 10, 2004. Therefore, the NRC staff provides results from the licensee's submittals and the NRC staff's review that relate to an assessment of whether the radiological consequences from the use of MOX LTAs on previously analyzed DBA would be expected to increase significantly.

The NRC staff's review focused on the potential impacts of the following three characteristics of MOX fuel: (1) the fission product inventory in a MOX fuel assembly 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 fuel assembly is greater due to the increased fission gas release (FGR) associated with higher fuel pellet centerline temperatures of MOX fuel, and (3) the increased FGR can result in higher fuel rod pressurization.

The configuration of the MOX LTAs is very similar to that of the LEU fuel assemblies currently in use at Catawba. No other plant modifications have been proposed by the licensee related to the use of MOX LTAs. 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 spent fuel pool decontamination factor due to higher fuel rod pressurization.

Radiological Consequence Analyses

Three categories of DBAs were analyzed for the effects of MOX LTAs. These accidents have been discussed in section 2.1 of this supplemental SE and will be summarized here. The first category of accidents involves damage to a significant portion of the entire core. They range in core damage from the LRA with an assumed 9.5 percent core damage for Unit 1 and 5.0 percent for Unit 2; the REA with an assumed 50 percent core damage, to the large break LOCA with full core damage. The results of Duke's analysis of these DBA categories are as follows:

For the LRA, the four MOX LTAs represent only 22 percent of the 18 affected assemblies for Unit 1 and 40 percent of the 10 affected assemblies for Unit 2. The potential increase in the iodine release and the thyroid dose is 14 percent for Unit 1 and 25.4 percent for Unit 2. The increase in exclusion area boundary (EAB) thyroid dose attributable to the use of MOX LTAs is from 23.6 to 26.9 rem for Unit 1 and from 22 to 27.8 rem for Unit 2.

For the REA, the four MOX LTAs represent only 4.1 percent of the affected 97 assemblies in the core. The potential increase in the iodine release and the thyroid dose is 2.62 percent. The increase in EAB thyroid dose attributable to the use of MOX LTAs is from 21.8 to 22.3 rem for Unit 1 and from 30.7 to 31.5 rem for Unit 2.

For the LOCA, the four MOX LTAs represent only 2.1 percent of the 193 assemblies in the core. The potential increase in the iodine release and the thyroid dose is 1.32 percent. The increase in EAB thyroid dose attributable to the use of MOX LTAs is from 89 to 90.2 rem for Units 1 and 2. The increase in thyroid dose to control room personnel attributable to the use of MOX LTAs is from 12.8 to 13 rem.

These changes in dose consequences that are attributable to the use of MOX LTAs result in doses that remain within the dose acceptance criteria, and do not represent a significant increase in the consequences of these previously evaluated accidents.

The second category of accidents includes the FHA, the WGD and the fresh MOX LTA drop accidents. 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 projected radiological consequences to increase for the FHA from 1.4 to 2.3 rem total effective dose equivalent (TEDE) at the EAB, from 0.21 to 0.34 rem TEDE at the outer boundary of the LPZ and from 1.3 to 2.1 rem TEDE in the control room. Duke projected radiological consequences for the WGD to increase from 2.2 to 3.5 rem TEDE at the EAB, from 0.31 to 0.5 rem TEDE at the outer boundary of the LPZ and from 2.1 to 3.3 rem TEDE in the control room.

Duke also assessed the radiological consequences of a drop of a fresh MOX LTA prior to it being placed in the spent fuel pool. 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. The EAB and control room TEDE estimated by the licensee for the postulated fresh fuel assembly drop were less than 0.3 rem. These consequences are bounded by the consequences of a dropped irradiated fuel assembly.

These resulting dose consequence values provide significant margin to the values specified in 10 CFR 50.67, “Accident Source Term,” as supplemented by regulatory position 4.4 of RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” and therefore, do not represent a significant increase in the consequences of these accidents.

The third category of accidents includes accidents whose source term assumptions are derived from RCS radionuclide concentrations. These include, steam generator tube rupture, main steam line 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.

II. New or Different Accident Evaluation

The proposed license amendment to allow the use of MOX LTAs will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The MOX LTAs have similar mechanical and thermal-hydraulic properties to, and nuclear characteristics only slightly different from, the current LEU fuel assemblies. The use of MOX LTAs does not involve any alterations to plant equipment or procedures that would introduce any new or unique operational modes or accident precursors. The existing design basis accidents have been evaluated to demonstrate that there is no significant adverse safety impact related to the use of MOX LTAs.

The main physical difference between a fresh MOX fuel assembly and a LEU fuel assembly is the presence of more radioactivity from the actinides in the MOX fuel matrix, resulting in a measurable dose rate in the immediate vicinity of a MOX fuel assembly. As a result, fresh MOX fuel is transported in a sealed leaktight shipping container by an enclosed tractor trailer truck. There are also differences in the fresh MOX fuel handling procedures, as described earlier in this SE, but these differences do not lead to a new or different type of accident.

A fuel handling accident involving a fresh MOX fuel assembly has potential for off-site dose consequences; however, the results of this fuel handling accident are bounded by the current analysis of a spent LEU fuel assembly drop accident. The calculated site boundary and control room dose consequences for a fresh MOX fuel handling accident are much less than the calculated doses for an accident involving a spent LEU fuel assembly and are well within the requirements in 10 CFR Part 100. This accident does not involve a new release path, does not result in a new fission product barrier failure mode, and does not create a new sequence of events that would result in significant cladding failure. Therefore, this accident is not a new or different kind of accident.

In conclusion, amending the Catawba license to allow the receipt, handling, storage, and use of 4 MOX LTAs does not create the possibility of a new or different kind of accident.

III. Margin of Safety Evaluation

The proposed license amendment to allow the use of MOX fuel lead assemblies will not involve a significant reduction in a margin of safety.

There are provisions in the Catawba TS that allow a "limited number of lead test assemblies" to be placed in "nonlimiting core regions." These provisions will not change and will apply to the planned use of MOX LTAs. The effect of these provisions is to place restrictions on the allowable power distribution limits for a MOX LTA.

The core design process assures that the limiting fuel rod in the core, whether LEU or MOX, has adequate nuclear power design limits under normal, transient, and accident conditions. If the core design process reveals unacceptable margin, adjustments are made to restore the needed margin. The operating limits are established in the Core Operating Limits Report to assure the design limits are not exceeded, thus assuring that adequate design margins for the fuel are maintained. This iterative design process is used to analyze the core containing MOX LTAs to assure that there is not a significant reduction in a margin of safety.

Because these LTAs will be located in nonlimiting locations i.e., will have margin above that of the limiting assemblies, the results of safety analyses will likewise assure that appropriate margins to safety are maintained during transients and accidents.

On the basis of the above evaluation, the NRC staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

4.0 UPDATING OF FINAL SAFETY ANALYSIS REPORT

The licensee stated in its letter dated February 27, 2003, that its application to amend the Catawba Facility Operating Licenses and TSs is not expected to require changes to the plants' UFSARs. The licensee further stated that if, as a result of implementing this license amendment, it determines that UFSAR changes are needed, appropriate changes will be made and submitted to the NRC in accordance with 10 CFR 50.71(e). The licensee restated this position in its submittal dated November 3, 2003, in response to the radiological consequences request for additional information number 1.

The NRC's regulations in 10 CFR 50.71(e) require that the UFSAR be updated to ". . . include the effects of: All changes made in the facility or procedures as described in the FSAR; all safety analyses and evaluations performed by the licensee [ ] in support of approved license amendments, . . . " The licensee has prepared numerous safety analyses that address both the MOX and NGF LTAs in support of its license amendment application to use four MOX LTAs at Catawba. That information is reflected in the submittals listed in the first paragraph of this Supplement to the SE and has been evaluated by the NRC staff as reported in the SE and its supplements. On this basis, the NRC staff concludes that the provisions of 10 CFR 50.71(e) require that the UFSAR for Catawba be updated to reflect the use of MOX and NGF LTAs.

5.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

On May 7, 2004, the Advisory Committee on Reactor Safeguards (ACRS) issued its letter, "Use of Mixed Oxide Lead Test Assemblies At The Catawba Nuclear Station" (ADAMS ML041320159). The ACRS letter noted consideration of the use of MOX fuel at the meeting of its Subcommittee on Reactor Fuels on April 21, 2004, and at the 512th meeting of the ACRS on May 5-8, 2004. The letter provided the ACRS's conclusion as follows: "We conclude that, under the restricted circumstances considered in both the Duke Power application and the NRC staff's safety evaluation, the four mixed oxide lead test assemblies in nonlimiting core locations that do not contain control rods can be irradiated in either of the Catawba reactor cores with no undue risk to the public health and safety."

6.0 COMMENTS ON PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Several comments were received in response to the NRC staff's July 12, 2004, proposed no significant hazards consideration (NSHC) determination (69 FR 41852). These comments and the NRC staff's response are grouped into subject area categories and addressed below.

6.1 Comment No. 1

Comment - Mr. Joe Whetstone of Bluffton, S.C., provided three comments that were received on July 22, 2004, (ADAMS ML042170079). The first comment referred to issues said to be before the Commodity Futures Trading Commission and stated that allowing Duke Energy Corporation to test plutonium fuel is a threat to our national security. The second comment stated that Duke had not addressed the issue of coolant blockage. The third comment was a general statement regarding security requirements for storing and transporting plutonium.

Response - The NRC staff's response to the first comment is that, with respect to the usage of MOX LTAs at Catawba, the issues said to be before the Commodity Futures Trading Commission are not within the regulatory purview of the NRC. With respect to the comment on national security, Duke has provided on page 5-5 of its License Amendment Request, the following statement from the Department of Energy:

> The fundamental purpose of the program is to ensure that plutonium produced for nuclear weapons and declared excess to national security needs (now and in the future) is never again used for nuclear weapons .....

In addition, the NRC staff notes that it has reviewed the augmented physical security provisions for the protection of the MOX fuel and has summarized its findings in Supplement No. 1, issued on May 5, 2004, to the NRC staff's original SE.

Regarding the second comment, the NRC staff notes that this issue has been addressed by an NRC Atomic Safety and Licensing Board (Board). On December 22, 2004, the Board issued its Partial Initial Decision wherein it found that a preponderance of the evidence showed that there is reasonable assurance that operation of Catawba with the four MOX LTAs will not endanger the health and safety of the public with respect to this issue. Accordingly, the NRC staff has no further comment.

Regarding the third comment, the NRC staff notes that numerous enhancements have been made to the provisions for nuclear power plant security since the September 11, 2001, events referenced by Mr. Whetstone.

6.2 Comment No. 2

Comment - Mr. Chad Simpson of Rock Hill, S.C. provide comments in an e-mail to the NRC staff dated August 10, 2004 (ADAMS ML042290017). Mr. Simpson requested that the comment period on the notice of a proposed finding of no significant hazards consideration that was published in the Federal Register on July 12, 2004, be extended. Mr. Simpson provided no specific basis for the request.

Response - In accordance with Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.91, the Federal Register notice provided a 30 day comment period for public comment on the NRC staff's proposed NSHC determination. The comment period expired on August 12, 2004. The NRC staff does not intend to extend the comment period for this notice.

7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

8.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact (EA) was published in the *Federal Register* on August 17, 2004 (69 FR 51112) (Reference 5). That EA also addressed exemptions from certain NRC regulations, issued by the NRC staff, that were dated March 3, 2005 (Reference 6). A Supplement to the Environmental Assessment and Finding of No Significant Impact was published in the *Federal Register* on February 23, 2005 (70 FR 8849) to address the concerns identified in Section 2.1 above. Accordingly, based on the EA and its supplement, the Commission has determined that the issuance of these amendments will not have a significant effect on the quality of the human environment. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, (69 FR 41852, dated July 12, 2004). The comments provided in response to this Notice and the NRC staff's responses, are discussed in Section 6.0 above.

9.0 CONCLUSION

The Commission has concluded, based on the considerations 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.

10.0 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," February 27, 2003.

2. Letter, R. E. Martin, NRC, to H. B. Barron, Duke, "Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies," April 5, 2004.

3. Letter, R. E. Martin, NRC, to H. B. Barron, Duke, "Supplement 1 to Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies," May 5, 2004.

4. Letter, R. E. Martin, NRC, to H. B. Barron, Duke, "Supplement 2 to Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies," dated July 27, 2004.

5. Letter, R. E. Martin, NRC, to H. B. Barron, Duke, "Catawba Nuclear Station, Units 1 and 2 - Environmental Assessment and Finding of No Significant Impact Related to the Use of Mixed Oxide Lead Test Assemblies," August 17, 2004.

6. Letter, R. E. Martin, NRC, to H. B. Barron, Duke, transmitting “Catawba Nuclear Station, Units 1 and 2 Re: Exemptions from Title 10 of the Code of Federal Regulations for the use of Mixed Oxide Fuel Lead Test Assemblies,” March 1, 2005.

7. Letter, W. R. McCollum, Duke, to NRC, “Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Dose Inputs),” August 31, 2004.

8. Letter, H. B. Barron, Duke, to NRC, “Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Revised Dose Evaluations),” September 20, 2004

9. Letter, R. E. Martin, NRC, to Duke, requesting additional information on radiological consequences, October 7, 2004.

10. Letter, H. B. Barron, Duke, to NRC, “Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Independent Review),” October 4, 2004.

11. Letter, H. B. Barron, Duke, to NRC, “Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Response to Request for Additional Information on Revised Dose Evaluations),” October 29, 2004.

12. Letter, W. R. Mc Collum, Jr., Duke, to NRC, “Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Additional Information on Revised Dose Evaluations),” December 10, 2004.

13. DPC-NE-3001P-A, “Duke Power Company, McGuire Nuclear Station, Catawba Nuclear Station, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology,” Duke Power Company, November 1991.

14. DPC-NE-3002-A, Revision 3, “UFSAR Chapter 15 System Transient Analysis Methodology,” May 1999.

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Date: March 3, 2005

TABLE 1

Catawba Atmospheric Dispersion Factors - χ/Q Values (sec/m^3)

Time (hr)	Receptor Location	
0–2 hours	Exclusion Area Boundary	4.78×10^{-4}
0–8 hours	Low Population Zone	6.85×10^{-5}
8–24 hours	Low Population Zone	4.00×10^{-5}
1–4 days	Low Population Zone	2.00×10^{-5}
4–30 days	Low Population Zone	7.35×10^{-6}

Control Room

Accident	0–8 hr	8–10 hr	10–24 hr	1–4 days	4–30 days
LRA, REA	9.9×10^{-4}	7.2×10^{-4}	7.2×10^{-4}	5.1×10^{-4}	2.8×10^{-4}
LOCA	1.98×10^{-3}	1.44×10^{-3}	7.2×10^{-4}	5.1×10^{-4}	2.8×10^{-4}

TABLE 2

ANALYSIS ASSUMPTIONS

Parameters Common to Rod Ejection and Locked Rotor Accidents

Core power (includes 2% uncertainty), MWt	3479
Iodine speciation fraction	
Elemental	0.91
Particulate	0.05
Organic	0.04
Iodine and noble gas gap fraction	0.10
Fraction of gap inventory released to primary coolant	1.0
Initial SG leak rate, gpd/SG	150
SG tube uncovery	Yes
Iodine partitioning (with SG tubes covered)	0.01
Control room volume, ft^3	89,200
Control room makeup flow rate, cfm	2667
Control room recirculation flow rate, cfm	1333
Control room recirculation filter efficiency	
Elemental	0.99
Particulate	0.99
Organic	0.95
Control room unfiltered inleakage, cfm	30

Locked Rotor Accident

Percent of core exceeding DNB		
Unit 1		9.5
Unit 2		5.0
Duration of plant cooldown, hrs		8
SG tube bundle uncovery, seconds	Unit 1	Unit 2
Steam Generator 1	1205	1870
Steam Generator 2	0	137
Steam Generator 3 and 4	1121	1770
Time-dependent SG tube leakage		Table QD8a-1 and QD8a-2[3]
Time-dependent SG mass		Table QD11-1 and QD11-2[3]
Time-dependent SG steam release		Table 1 of Attachment 2[3]

[3] October 29, 2004 RAI response.

Rod Ejection Accident

Fraction of core exceeding DNB			0.5
Fraction of core that is melted			0
Fraction of gap inventory released to containment			
Noble gases			1.0
Iodine			0.25
Fraction of gap inventory released to containment sump			0.5
Fraction of gap inventory released to primary coolant			1.0
Containment volume, ft^3			1,196,000
Containment leak rate, %/day	Collected	Bypass	Total
Prior to annulus drawdown	0	0.3	0.3
Drawdown to 24 hours	0.279	0.021	0.3
After 24 hours	0.1395	0.0105	0.15
Annulus drawdown, sec			95
Annulus volume, ft^3 (assuming 50% credit)			242,000
Annulus Ventilation flow rate, cfm			8100
Containment air return fan (after 600 seconds), cfm			40,000
Elemental iodine ice condenser removal efficiency, %			30
Containment spray credit			none
Containment sump volume, ft^3			79,000
ECCS leakage rate (includes 2x multiplier), cc/hr			7520
ECCS leakage iodine partitioning			0.1
Duration of plant cooldown, hrs			8
SG tube bundle uncovery, seconds		Unit 1	Unit 2
Steam Generator 1		751	2425
Steam Generator 2		662	2605
Steam Generator 3 and 4		2458	810
Time-dependent SG tube leakage			Table QD8a-3 and QD8a-4[4]
Time-dependent SG mass			Table QD11-3 and QD11-4[4]
Time-dependent SG steam release			Table 1 of Attachment 2[4]

[4] October 29, 2004 RAI response

Control Room Dose from LOCA

Core power (includes 2% uncertainty), MWt			3479
Fraction of fission product activity available for release			
Noble gases to containment atmosphere			1.0
Iodine to containment atmosphere			0.25
Iodine to containment sump			0.5
Iodine speciation fraction			
Elemental			0.91
Particulate			0.05
Organic			0.04
Containment volumes, ft^3			
Total			1,015,400
Upper			670,000
Lower			345,000
Sump			79,000

Containment Leakage Pathway

Containment leak rate, %/day	Collected	Bypass	Total
Prior to annulus drawdown	0	0.3	0.3
Drawdown to 24 hours	0.279	0.021	0.3
After 24 hours	0.1395	0.0105	0.15

Annulus drawdown, sec	95
Annulus volume, ft^3 (assuming 50% credit)	242,000
Annulus ventilation flow rate, cfm	Table 1 of Attachment 2[5]
Annulus filtration efficiency, %	95
Containment air return fan (after 600 seconds), cfm	40,000
Elemental iodine ice condenser removal efficiency, %	30
Containment spray credit, hr^{-1}	none
Elemental	0.9
Particulate	2.4
Organic	0

[5] October 29, 2004 RAI response

Control Room Parameters

Control room volume, ft^3	89,200
Control room makeup flow rate, cfm	2800
Control room recirculation flow rate, cfm	2000
Control room recirculation filter efficiency	
Elemental	0.99
Particulate	0.99
Organic	0.95
Control room unfiltered inleakage, cfm	30

ECCS Leakage Pathway

ECCS leakage rate (includes 2x multiplier), cc/hr	7520
ECCS leakage iodine partitioning	0.1
ECCS recirculation start, sec	1700

TABLE 3

April 5, 2004 Safety Evaluation Section 3.2.5 At-Power Core Damage Accidents	Updated Evaluation
Licensee's Analysis[6]	Licensee's Analysis
For the LOCA, the four MOX LTAs represent 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 resulting doses are shown in parentheses in the first column of Table 4.	For the LOCA, the four MOX LTAs represent 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 resulting doses are shown in the second column of Table 4.
For the LRA, the four MOX LTAs represent 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 resulting doses are shown in parentheses in the first column of Table 4.	For the LRA at Unit 1, the four MOX LTAs represent 22 percent of the 18 affected assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 14.1 percent for Unit 1. For the LRA at Unit 2, the four MOX LTAs represent 40 percent of the 10 affected assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 25.4 percent for Unit 2. The resulting doses are shown in the second column of Table 4.
For the REA, the four MOX LTAs represent 4.1 percent of the affected 97 assemblies in the core. Thus, the potential increase in the iodine release and the thyroid dose is 2.62 percent. The resulting doses are shown in parentheses in the first column of Table 4.	For the REA, the four MOX LTAs represent only 4.1 percent of the 97 affected in the core. Thus, the potential increase in the iodine release and the thyroid dose is 2.62 percent. The resulting doses are shown in the second column of Table 4.
NRC Staff's Analysis[6]	NRC Staff's Analysis
For the LOCA, the potential increase in the iodine release and the resulting thyroid dose is 1.53 percent.	For the LOCA, the potential increase in the iodine release and the resulting thyroid dose is 1.53 percent.

[6] The licensee's analysis is based on multipliers for MOX fuel of 1.09 for Iodine-131 and 1.5 for fission gas and the NRC staff's analysis is based on multipliers for MOX fuel of 1.158 for Iodine-131 and 1.5 for fission gas as discussed in Section 3.2.5 of Reference 2.

For the LRA, the potential increase in the iodine release and the resulting thyroid dose is 14.0 percent.	For the LRA at Unit 1, the potential increase in the iodine release and the resulting thyroid dose is 16.4 percent as compared to 14.1 percent determined by Duke. For the LRA at Unit 2, the potential increase in the iodine release and the resulting thyroid dose is 29.5 percent as compared to 25.4 percent determined by Duke.
For the REA, the potential increase in the iodine release and the resulting thyroid dose is 3.04 percent.	For the REA, the potential increase in the iodine release and the resulting thyroid dose is 3.04 percent as compared to 2.62 percent determined by Duke.

TABLE 4
DESIGN BASIS ACCIDENT DOSES BY LICENSEE

	All LEU Core[7]	LEU Core Plus 4-MOX LTAs	Acceptance Criteria
LOCA, *rem Thyroid*			
EAB	(89) 89	90.2	300
LPZ	(25) 12.7	12.9	300
Control Room	(5.3) 12.8	13.0	30
U1 Locked Rotor Accident, *rem Thyroid*			
EAB	(3.7) 23.6	26.9	30
LPZ	(1.2) 4.1	4.6	30
Control Room	(NA) 0.9	1.0	30
U2 Locked Rotor Accident, *rem Thyroid*			
EAB	(3.7) 22.0	27.8	30
LPZ	(1.2) 3.6	4.5	30
Control Room	(NA) 1.1	1.4	30
U1 Rod Ejection Accident, *rem Thyroid*			
EAB	(1.0) 21.8	22.3	75
LPZ	(0.1) 17.4	17.8	75
Control Room	(NA) 6.4	6.6	30
U2 Rod Ejection Accident, *rem Thyroid*			
EAB	(1.0) 30.7	31.5	75
LPZ	(0.1) 19.3	19.8	75
Control Room	(NA) 8.7	8.9	30

Weir Gate Drop Accident
Fuel Handling Accident
Fresh LTA Drop

No change from the values reported in Table 4 of Reference 2.

[7] Values in parentheses are the "All LEU Core" values from Table 4 of Reference 2 SE.

7590-01-P

UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE ENERGY CORPORATION, ET AL.

DOCKET NOS. 50-413 AND 50-414

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

RENEWED FACILITY OPERATING LICENSE NOS NPF-35 NPF-35

AMENDMENT NOS. 220 AND 215

The Nuclear Regulatory Commission (Commission) has issued Amendment No. 220 to Renewed Facility Operating License No. NPF-35 and Amendment No. 215 to Renewed Facility Operating License No. NPF-52, issued to Duke Energy Corporation, et al. (Duke, the licensee), which revised the Technical Specifications (TS) for operation of the Catawba Nuclear Station (Catawba), Units 1 and 2, located in York County, South Carolina. The amendment is effective as of the date of issuance.

The amendment modifies the TS to permit the usage of up to four mixed oxide (MOX) lead test assemblies (LTAs). Specifically, the amendment consists of: 1) a revision to TS 3.7.16 to permit storage of the MOX LTAs in the spent fuel pool; 2) a revision to TS 4.2, “Reactor Core” to include the four MOX LTAs using M5 fuel rod cladding; 3) TS 4.3, “Fuel Storage,” to reflect the enrichment of the MOX LTAs; and 4) a revision to TS 5.6.5 to add two supporting methodologies for the MOX LTAs. The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission’s rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

A Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the *FEDERAL REGISTER* on July 25, 2003 (68 FR 44107). A request for a hearing was filed on August 21 and 25, 2003, by the Nuclear Information and Resources Service (NIRS) and the Blue Ridge Environmental Defense League (BREDL), respectively. A Notice of Opportunity to Comment and Proposed No Significant Hazards Consideration Determination in connection with this action was published in the *FEDERAL REGISTER* on July 12, 2004 (69 FR 41852).

On July 14 and 15, 2004, the Atomic Safety and Licensing Board (ASLB) held a hearing on a single admitted safety-related contention by BREDL. All of NIRS's contentions were rejected and NIRS was not admitted as a party to the proceeding. The admitted contention was related to the adequacy of the loss-of-coolant accident analyses performed to support the use of the MOX LTAs. On December 22, 2004, the ASLB issued a Partial Initial Decision with respect to this matter finding that there is reasonable assurance that operation of Catawba with the four MOX LTAs will not endanger the health and safety of the public.

BREDL submitted its security-related safety contentions on March 3, 2004. An ASLB hearing on a single physical security-related contention, as admitted by the ASLB, was held January 11-14, 2005. This contention was related to the adequacy of the provisions undertaken by Duke to provide protection of the MOX LTAs. Findings and reply findings of fact and conclusions of law were filed in February 2005. An ASLB decision on the security contention is pending.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding or completion of any required hearing, where it has determined that no significant hazards considerations are involved.

The Commission has applied the standards of Title 10 of the *Code of Federal Regulations*, Section 50.92 and has made a final determination that the amendment involves no significant hazards considerations. The basis for this determination is contained in a Safety Evaluation and three Supplements to that Safety Evaluation related to this action. Accordingly, as described above, the amendment has been issued and made immediately effective and any further hearing will be held after issuance.

The Commission has prepared an Environmental Assessment and one Supplement to the Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment and its supplement, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (69 FR 51112 and 70 FR 8849).

For further details with respect to the action see (1) the application for amendment dated February 27, 2003, as supplemented by letters dated September 15, September 23, October 1 (two letters), October 3 (two letters), November 3, November 4, December 10, 2003, and February 2, (two letters), March 1 (three letters), March 9 (two letters), March 16 (two letters), March 26, March 31, April 13, April 16, May 13, June 17, August 31, September 20, October 4, October 29 and December 10, 2004, (2) Amendment Nos. 220 and 215 to License Nos. NPF-35 and NPF-52, respectively, (3) the Commission's related Safety Evaluation and its three Supplements dated April 5, May 5, July 27, 2004, and March 3, 2005, respectively, and (4) the Commission's Environmental Assessment and its supplement (69 FR 51112 and 70 FR 8849, respectively). All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, File Public Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 3rd day of March 2005.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 3rd day of March 2005.

FOR THE NUCLEAR REGULATORY COMMISSION

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