March 17, 2005

G. R. Peterson, Vice President
McGuire Nuclear Station
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MC0945 AND MC0946)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 225 to Renewed Facility Operating License NPF-9 and Amendment No. 207 to Renewed Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated September 29, 2003, as supplemented by letters dated April 22, May 20, June 9, and July 29, 2004.

The amendments revise the spent fuel pool storage criteria based upon fuel type, fuel enrichment, burnup, cooling time and partial credit for soluble boron. These amendments also allow for the safe storage of fuel assemblies with a nominal enrichment of Uranium isotope (U-235) up to 5.00 weight percent.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

James J. Shea, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 225 to NPF-9
2. Amendment No. 207 to NPF-17
3. Safety Evaluation

cc w/encls: See next page

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OFFICIAL RECORD

<u>DUKE ENERGY CORPORATION</u>

<u>DOCKET NO. 50-369</u>

<u>MCGUIRE NUCLEAR STATION, UNIT 1</u>

<u>AMENDMENT TO RENEWED FACILITY OPERATING LICENSE</u>

Amendment No. 225
Renewed License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. NPF-9 filed by the Duke Energy Corporation (licensee) dated September 29, 2003, as supplemented by letters dated April 22, May 20, June 9, and July 29, 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-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 225, are hereby incorporated into this license. The licensee 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 120 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 17, 2005

DUKE ENERGY CORPORATION

DOCKET NO. 50-370

MCGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 207
Renewed License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. NPF-9 filed by the Duke Energy Corporation (licensee) dated September 29, 2003, as supplemented by letters dated April 22, May 20, June 9, and July 29, 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-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 207, are hereby incorporated into this license. The licensee 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 120 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 17, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 225

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND LICENSE AMENDMENT NO. 207

RENEWED FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

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

Remove	Insert
3.7.15-1	3.7.15-1
thru	thru
3.7.15-21	3.7.15-32
4.0-1	4.0-1
thru	thru
4.0-2	4.0-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 225 TO RENEWED FACILITY

OPERATING LICENSE NPF-9 AND

AMENDMENT NO. 207 TO RENEWED FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC or the Commission) dated September 29, 2003, as supplemented by letters dated April 22, May 20, June 9, and July 29, 2004, Duke Energy Corporation (Duke, the licensee), submitted a request for changes to the McGuire Nuclear Station (McGuire), Units 1 and 2, Technical Specifications (TSs). The requested changes would revise the spent fuel pool (SFP) storage criteria based upon fuel type, fuel enrichment, burnup, cooling time and partial credit for soluble boron, TSs 3.7.15 and 4.3. Upon approval of these changes, fuel assemblies with a nominal enrichment of U-235 up to 5.00 weight percent will be allowed and the required soluble boron credit will decrease to 800 ppm. In addition, the request assumes full credit for the Boral® neutron poison material in Region 1 of the SFP and no credit for the degrading Boraflex neutron poison material in Region 2. Region 1 of the SFP was reracked with new replacement in-kind fuel racks utilizing Boral® neutron poison in place of Boraflex in accordance with the stipulations as set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59. The supplemental letters dated April 22, May 20, June 9, and July 29, 2004, provided additional information requested by the NRC staff on the expected Boral® material performance and testing, and to clarify as needed the criticality analysis that was provided in the licensee submittal.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act of 1954, as amended (the "Act") requires applicants for nuclear power plant operating licenses to include TSs as a part of the license. The TSs ensure the operational capability of structures, systems and components that are required to protect the health and safety of the public. The NRC's regulatory requirements that are related to the content of the TSs are contained in 10 CFR 50.36. 10 CFR 50.36 requires that the TSs include items in the following specific categories: (1) safety limits, limiting safety system settings, and limiting control settings (50.36(c)(1)); (2) limiting conditions for operation (LCOs) (50.36(c)(2)); (3) surveillance requirements (SRs) (50.36(c)(3)); (4) design features (50.36(c)(4)); and (5) administrative controls (50.36(c)(5)).

Pursuant to 10 CFR 50.90, a licensee may apply for an amendment to its license, including the TSs incorporated into the license. In determining the acceptability of the proposed changes, the NRC staff interprets the requirements of the current version of 10 CFR 50.36. Within this general framework, licensees may revise their current TSs provided that a plant-specific review supports a finding of continued adequate safety whereas: (1) the change is editorial, administrative, or produces clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but continues to afford adequate assurance of safety when judged against current regulatory standards.

2.1 Spent Fuel Pool Regulatory Requirements

10 CFR Part 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," provides a list of the minimum design requirements. According to GDC 62, "Prevention of criticality in fuel storage and handling," the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes.

10 CFR 50.68, "Criticality accident requirements," provides the NRC regulations for maintaining subcritical conditions in SFPs. As part of its proposed changes to the design and operation of the SFP, the licensee is also changing the licensing basis of the spent fuel pool from an exemption to 10 CFR 70.24, "Criticality accident requirements," to compliance with 10 CFR 50.68. As such, the NRC staff reviewed the licensee's compliance with all the requirements of 10 CFR 50.68.

The 10 CFR 50.68 acceptance criteria for criticality prevention in the SFP that are applicable to the licensee's proposed amendment are the following:

1. the effective multiplication factor (k_{eff}) shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95 percent probability, 95 percent confidence (95/95) level; and

2. k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

Additionally, the licensee proposes in its amendment request to change the licensing basis of the new fuel vault. The 10 CFR 50.68 acceptance criteria for criticality prevention in the new fuel vault that are applicable to the licensee's proposed amendment are the following:

1. k_{eff} of the fresh fuel in the new fuel vault shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95 at a 95/95 level; and

2. k_{eff} of the fresh fuel in the new fuel vault shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and filled with a low-density hydrogenous fluid, resulting in optimum moderation conditions, and must not exceed 0.98 at a 95/95 level.

2.2 NRC Acceptable Criticality Methodologies

The NRC defined acceptable methodologies for performing SFP criticality analyses in three documents:

1. NUREG-0800, Standard Review Plan, Section 9.1.2, "Spent Fuel Storage," Draft Revision 4;

2. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis,"; and

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

The NRC staff used the guidance contained in these documents to assist in its review of the licensee's amendment request.

The NRC staff reviews the compatibility and chemical stability of the poison material in the SFP environment in accordance with Standard Review Plan (SRP) 9.1.2, "Spent Fuel Storage," to ensure that there are no potential mechanisms that will: (1) alter the dispersion of the strong fixed neutron absorbers incorporated in the design of the storage racks, and/or (2) cause physical distortion of the tubes retaining the stored fuel assemblies. In addition, the NRC staff reviews any commitments related to the verification of the compatibility and chemical stability of the poison materials used in the SFP storage racks.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment. The evaluation below will support the conclusion that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

3.1 Criticality Analysis

In determining the acceptability of Duke's amendment request, the NRC staff reviewed three aspects of the licensee's analyses: 1) the computer codes employed; 2) the methodology used to calculate the maximum k_{eff}; and 3) the storage configurations and limitations proposed. For each part of the review the NRC staff evaluated whether the licensee's analyses and methodologies provided reasonable assurance that adequate safety margins in accordance with NRC regulations were developed and can be maintained in the McGuire, Units 1 and 2, SFPs.

3.1.1 Computer Codes

The licensee performed the analysis of the reactivity effects for the McGuire, Units 1 and 2, SFP racks with the KENO V.a code (KENO), a three-dimensional Monte Carlo criticality code. KENO was benchmarked against criticality experiments under conditions that considered the effects of varying fuel enrichment as well as both over and under-moderated lattices. The experimental data are sufficiently diverse to establish that the method bias and uncertainty will apply to McGuire, Units 1 and 2 storage rack conditions. The licensee determined the KENO calculation (methodology) bias is 0.0064 with a 95/95 bias uncertainty of +/- 0.0066 using the 238-group cross section library for the SFP storage racks. Duke also calculated a benchmark calculation (methodology) bias of 0.0061 with a 95/95 bias uncertainty of +/- 0.0071 for the new fuel vault (NFV) storage racks.

In addition to using KENO to perform the criticality analyses, the licensee employed the CASMO-3/SIMULATE-3 code set to perform the fuel depletion analyses and the Region 2 irradiated-fuel criticality analyses. The licensee used this two-dimensional multi-group transport theory code to determine the isotopic composition of the spent fuel. Additionally, the licensee determined the reactivity effect (delta-k) for each manufacturing tolerance of the fuel assemblies and storage racks. Duke used the fine-energy-group (70-group) neutron cross-section library and determined the critical benchmark calculation bias was -0.0015 with a 95/95 bias uncertainty of +/- 0.0121 for its CASMO-3 analyses. The licensee, in accordance with NRC guidance documents, conservatively ignored the negative calculation bias in its criticality analyses

The NRC staff reviewed the licensee's application of the codes to determine whether each could reasonably calculate the appropriate parameters necessary to support the maximum k_{eff} analyses. In the NRC memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," dated August 19, 1998, the NRC staff stated that KENO and CASMO were acceptable computer codes for the analysis of fuel assemblies stored in the SFP. Additionally, in accordance with an NRC letter dated August 19, 1998, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," the NRC staff stated that the Babcock and Wilcox series of criticality experiments, "BAW-1487," July 1979 and "BAW-1645-4," 1981, provided an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control and close-packed arrays of fuel. Therefore, the NRC staff concludes that the licensee's use of KENO for calculation of the nominal k_{eff} was appropriate since it was benchmarked against experimental data that reflects the proposed assembly and rack conditions for the McGuire, Units 1 and 2, SFP. Additionally, the NRC staff finds that the licensee's use of the CASMO-3/SIMULATE-3 code set was acceptable for determining the delta-k for each manufacturing tolerance, performing the fuel depletion analyses, and calculating the nominal k_{eff} for the Region 2 racks.

3.1.2 Methodology

The licensee performed criticality analyses of its SFP in accordance with the guidance contained in the three NRC defined acceptable methodologies listed in paragraph 2.2 of this Safety Evaluation (SE). The licensee combined a worst-case analysis based on the bounding fuel and rack conditions, with a sensitivity study using 95/95 analysis techniques. The major components in this analysis were a calculated k_{eff} based on the limiting fuel assembly, SFP temperature and code biases, and a statistical sum of 95/95 uncertainties and worst-case delta-k manufacturing tolerances.

In performing its criticality analysis, the licensee first calculated a k_{eff} based on nominal core conditions using either the KENO or CASMO-3 code. Duke calculated this nominal k_{eff} for every fuel assembly design, fuel enrichment, cooling time, and storage region combination that is considered in the scope of the McGuire, Units 1 and 2, SFP and NFV criticality analyses. Additionally, the licensee included the effects of the bounding SFP temperatures in the determination of the nominal k_{eff}. The licensee used the minimum and maximum permissible SFP temperatures and corresponding water densities to determine which condition resulted in the most limiting nominal k_{eff}.

To the calculated k_{eff}, the licensee added the methodology bias as well as additional biases such as the fixed poison self-shielding bias and the cooling time/enrichment interpolation error. As stated in the description of the KENO and CASMO-3 codes, the licensee determined the methodology bias from the critical benchmark experiments. Duke only applied the fixed poison self-shielding bias to criticality analyses performed for Region 1 since only Region 1 contains a credited fixed neutron absorbing poison, Boral®. Likewise, Duke applied the cooling time/enrichment interpolation error only to Region 2 criticality analyses because the Region 1 analyses do not take credit for burnup and cooling time. The fixed poison self-shielding bias accounts for the slight self-shielding effects caused by the clustering of boron carbide particles within the Boral® panels. The licensee determined the maximum bias possible from all the evaluated variations and combinations of temperature and soluble boron concentrations. Duke conservatively added the maximum fixed poison self-shielding bias to all nominal Region 1 k_{eff} cases it calculated. In addition to the fixed poison self-shielding bias, the licensee included a cooling time/enrichment interpolation error (bias) to account for the maximum difference in k_{eff} between a minimum burnup limit estimated using the interpolation technique and actual burnup limit that a specific evaluation at that enrichment and cooling time would yield. Duke quantified the error by performing analyses of various intermediate enrichments and/or cooling times and compared the results with the interpolation estimates. For example, the licensee performed its intermediate analysis for the Region 2 unrestricted storage cases at both 4.75 weight percent and 5.0 weight percent enrichments and cooling time k_{eff}s computed at 2.5-year intervals. These intermediate analyses gave the licensee the additional information it needed to calculate the interpolation error. The licensee identified the maximum interpolation error from all the cases it calculated and applied it as a bias to the nominal k_{eff} results in all Region 2 analyses.

For added conservatism the licensee neither extrapolates below fuel assembly enrichments nor above the 20-year cooling times provided in the proposed TS Tables 3.7.15-1 through 3.7.15-4. For example, a fuel assembly enriched to less than the limits provided in its respective tables is assumed by the licensee to be enriched to the minimum limit provided in the table. Likewise, fuel assemblies that have cooled for more than 20 years are only credited to the 20-year limit in the tables. Both of these assumptions ensure that the licensee takes appropriate and conservative credits based on the analyses it has performed in support of its amendment request.

Additionally, to determine the maximum k_{eff}, the licensee performed a statistical combination of the reactivity effects for code and methodology uncertainties, manufacturing tolerances, and burnup uncertainties. The code and methodology uncertainties included the benchmark method uncertainty and the Monte Carlo computational uncertainty. The licensee determined both of these uncertainties to a 95/95 threshold that is consistent with the requirements of 10 CFR 50.68. In addition to including the code and methodology uncertainties, the licensee performed detailed analyses to determine appropriate and conservative mechanical tolerances as well as a burnup computational uncertainty, a burnup measurement uncertainty, and an axial profile uncertainty. In response to a request for additional information (RAI) dated April 22, 2004, the licensee provided, a comprehensive list of the manufacturing tolerances considered as well as the reactivity effect calculated for each. For each tolerance, the licensee calculated a delta-k between the nominal condition and the most limiting tolerance condition. By using the most limiting tolerance condition, the licensee calculated the highest reactivity effect possible. This results in a conservative margin since the tolerances will always bound the actual parameters.

The licensee places considerable emphasis in its criticality analyses on a burnup credit; therefore, the accurate determination of burnup uncertainties is essential to ensure the regulatory limits on k_{eff} are satisfied. In its criticality analyses, the licensee included the effects of the burnup computational uncertainty, burnup measurement uncertainty, and axial profile uncertainty.

The burnup computational uncertainty quantifies the ability of the CASMO-3/SIMULATE-3 codes to accurately determine the isotopic content of a collection of irradiated assemblies in the McGuire, Units 1 and 2, reactors, assuming the actual average burnup of the fuel in the reactor core is the same as the average burnup of the SIMULATE model for the reactor core. To determine the uncertainty, the licensee analyzed several cycles of McGuire, Units 1 and 2, reactor operational data to evaluate the differences between measured and SIMULATE-3 predicted core reactivity at various times during the operating cycle. The licensee's analysis considered the effects of variations in the five pertinent irradiation history parameters: 1) axial burnup, 2) moderator temperature, 3) fuel temperature, 4) soluble boron concentration, and 5) burnable poison exposure. The licensee's analysis used all the available core-follow data for the three-dimensional values of these variables in the actual discharged fuel assemblies. The licensee determined the burnup computational uncertainty based on actual core operational data and developed a bounding linear representation of the uncertainty as a function of the individual fuel assembly burnup.

The licensee also determined the burnup measurement uncertainty that represents the reactivity penalty associated with the difference between the measured burnup and the code-predicted burnup. Determination of the measured burnup, which is used for the TS verification of the minimum burnup requirements, introduces multiple sources of instrument error that results in the measured burnup being different from the actual burnup of the fuel assembly. The licensee used the measured burnups contained in its master special nuclear material database, as obtained from the in-core detector measurements taken regularly during power operation, and compared these values to the predicted burnups taken from reactor core-follow computations. The licensee compared the predicted-to-measured burnups of more than 1900 fuel assemblies and computed and sorted the distribution of these differences. The licensee then compared this distribution to a randomly-generated normal distribution of the same sample size and same standard deviation. As described in their response to a request for additional information (RAI) dated April 22, 2004, the results of this comparison showed that the calculated and randomly-generated distributions correlated very well. The licensee then used the distribution of predicted-to-measured errors to quantify the largest conceivable average burnup error for various randomly-selected clustered arrays of fuel assemblies. From these arrays, the licensee determined the highest "average" error and applied it to model that array with fuel at a different burnup from the surrounding fuel assemblies. The licensee then identified the largest system k_{eff} increase observed in each of the models, among all array sizes evaluated, and conservatively applied it as the bounding burnup measurement uncertainty.

For the axial profile uncertainty, the licensee followed the same statistical approach it used in determining the bounding burnup measurement uncertainty. The licensee used the axial profile uncertainty to establish the bounding reactivity penalty associated with differences between the k_{eff} calculated using the average "estimated" axial burnup and history profiles for a particular fuel assembly, and the k_{eff} calculated using the actual axial burnup and history profiles for that fuel assembly. The licensee followed a similar approach in the selection of the arrays used in the axial profile uncertainty determination as it did in the burnup measurement uncertainty determination. However, for the axial profile uncertainty determination, the licensee was able to group assemblies in arrays based on those that had been symmetrically designed for reactor operation since these assemblies will have the same axial profile characteristics when discharged from the reactor together. Again, the licensee determined the highest "average" error for these arrays and applied it to model that array with fuel at a different axial history profile from the surrounding fuel assemblies. The licensee identified the largest system k_{eff} increase observed in each of these models, among all array sizes evaluated, and conservatively applied it as the bounding axial profile uncertainty.

Once the reactivity effects for each of the tolerances and uncertainties were determined, the licensee statistically combined these results in accordance with the guidance contained in an NRC letter dated August 19, 1998, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants.” At the NRC staff's request, the licensee provided additional information in an April 22, 2004, RAI response, depicting the statistical methodologies employed for combining these tolerances and uncertainties. The NRC staff reviewed the licensee’s methodology for calculating each of the reactivity effects associated with uncertainties and manufacturing tolerances as well as the statistical methods used to combine these values. The NRC staff finds the licensee’s methods for calculating the maximum k_{eff} conservative and acceptable.

Since the Region 2 racks are of an egg-crate design, the licensee homogenized the Region 2 rack model to simplify its analysis. In its amendment request, the licensee described how it homogenized the Region 2 rack model for analysis using CASMO-3. The licensee stated that the cell wall location was adjusted in the model to be located at the midpoint between the stored assemblies, thereby, making neighboring cells identical to each other. However, this change affects the amount of moderator directly adjacent to each assembly causing differences between the maximum k_{eff} observed in the model and what exists in the actual Region 2 SFP storage racks. At the NRC staff's request, the licensee provided detailed information in a May 20, 2004, RAI response to demonstrate that the homogenized Region 2 racks modeled with CASMO-3 bounded both the as-built (heterogeneous) racks modeled with KENO and the homogenized Region 2 racks modeled with KENO. To demonstrate that its CASMO-3 homogeneous model was bounding, the licensee performed a number of analyses for each model and code in which it varied fuel assembly enrichments and types as well as SFP temperature. The licensee's results show that the CASMO-3 homogenized rack model used by the licensee provides slightly conservative and bounding results when compared to KENO heterogeneous and homogenized Region 2 rack models. Therefore, the NRC staff finds that the model used by the licensee to analyze the Region 2 racks is conservative and acceptable.

3.1.3 Proposed Storage Configurations

The primary purpose of the licensee's amendment request was to gain the NRC staff's approval for new storage configurations within the SFP storage racks. The licensee's proposed TS LCO 3.7.15a would permit unrestricted storage of new or irradiated fuel in Region 1 of the McGuire, Units 1 and 2, SFP provided the maximum initial enrichment was less than or equal to 5.0 weight percent. The licensee's proposed Region 2 storage configurations are divided into two permissible checkerboard (2 by 2) storage patterns based on four fuel assembly storage classifications. TS Figures 3.7.15-1 and 3.7.15-2 depict the acceptable storage patterns and provide the limitations for each. The first pattern, as shown in TS Figure 3.7.15-1, consists of a 2 by 2 pattern made up of fuel assemblies classified as either restricted or filler fuel assemblies. The second pattern, as show in TS Figure 3.7.15-2, consists of a 3 of 4 pattern wherein 3 cells contain fuel assemblies classified as checkerboard fuel while the remaining cell remains empty. Additionally, the figures describe the allowable rack interface alignments, wall-interface storage patterns, and inter-rack storage requirements.

The licensee's proposed storage patterns require proper grouping of spent fuel assemblies into the four storage classifications based on initial enrichment, burnup, and cooling time. The four fuel assembly storage classifications, in order of increased restrictions on minimum burnup and/or cooling time, are 1) unrestricted storage, 2) checkerboard storage, 3) restricted storage, and 4) filler storage. TS Tables 3.7.15-1 through 3.7.15-4 provide numerical data used to calculate the minimum burnup as a function of initial enrichment and cooling time. The licensee classifies each fuel type based on its ability to meet the three requirements (enrichment, burnup, and cooling time). To demonstrate the acceptability of the data presented in the tables, the licensee performed numerous confirmatory calculations based on the tabular values. The results showed that the k_{eff} was less than 1.0 in all cases without crediting soluble boron. Additionally, the confirmatory calculations, in combination with the conservative assumptions, ensure that the results bound the actual variance in conditions found in the SFP.

In addition to classifying the assemblies based on their fuel depletion characteristics, the licensee also had to consider the effects of Burnable Poison Rod Assemblies (BPRAs) used in some of the fuel types at McGuire, Units 1 and 2. Higher BPRA boron concentrations yield higher k_{eff} increases in the fuel assemblies that contained the BPRAs during irradiation. Duke only analyzed the effect of the BPRAs in the Region 2 analyses because the Region 1 analyses assume unirradiated assemblies at the maximum permissible enrichment. The presence of BPRA rodlets in an irradiated fuel assembly increase the production of fissile plutonium because each displaces the moderator in the fuel assembly lattice, resulting in local spectral hardening. This results in increased residual reactivity in spent fuel assemblies. For conservatism, the licensee assumed that the maximum number of BPRA rodlets were loaded into assemblies that underwent irradiation with BPRAs present. Therefore, the licensee's analyses for assemblies loaded with BPRAs will be bounding. Additionally, in its original amendment request, the licensee assumed BPRA boron carbide concentrations "at or very near" the highest boron concentrations that were used in the BPRAs for the respective fuel assembly types. For example, the licensee assumed that the MkBI[1] BPRA contained 1.4 weight percent boron carbide. However, the maximum boron carbide concentration used in the MkBI[1] BPRA was 1.43 weight percent. At the NRC staff's request the licensee performed analyses that showed that the small difference in concentration resulted in a negligible increase in the residual reactivity of the affected fuel assemblies. The licensee also evaluated the reactivity effect of mechanical tolerances in the design and manufacture of the BPRAs. Duke provided sufficient information to demonstrate that the effects of these variations were negligible. Therefore, the NRC staff finds that the licensee's analyses of the residual reactivity for irradiated fuel assemblies that contained BPRAs is conservative and acceptable.

Since the licensee places considerable emphasis on burnup credit in its SFP criticality analyses, the NRC staff requested the licensee provide additional information demonstrating proper controls existed to ensure the minimum burnup limits were met. In additional information the licensee provided on April 22, 2004, the licensee stated that the controls used for determining the actual spent fuel assembly burnups were currently in place for determining burnup for existing spent fuel pool requirements as well as fuel reload analyses. The licensee determines the burnup and isotopic weights for each fuel assembly based on flux maps taken during cycle operation. Following the end of a cycle, the licensee determines the final burnup of each assembly and records the information in its special nuclear material database. As described in supplemental RAI responses dated April 22 and May 20, 2004, the licensee performs its flux maps and determines its fuel assembly burnups in accordance with NRC-approved methodologies.

In addition to crediting fuel assembly burnup, the licensee has proposed to credit the cooling time for fuel assemblies. The licensee’s cooling time credit accounts for the decay of longer-life nuclides such as plutonium-241 as well as the buildup of poisons such as gadolinium-155. The licensee uses the CASMO-3 code to calculate the reactivity credit available as a function of the decay of these nuclides.

The licensee calculated maximum k_{eff} values for each of the proposed SFP storage cases. The licensee’s results show the maximum k_{eff}s of 0.9829 for SFP Region 1 storage and 0.9989 for SFP Region 2 storage for unborated cases. Additionally, the licensee calculated the required soluble boron concentration under normal conditions that would yield a maximum k_{eff} of 0.95. The analysis determined the required concentration to meet this limit was 800 ppm. This 800 ppm concentration, which is included in the licensee's proposed TS 4.3.1.1c, is well below the TS limit of 2675 ppm described in the McGuire, Units 1 and 2, Core Operating Limits Report. In addition to performing the criticality analyses for normal storage conditions, the licensee analyzed accident conditions such as a mislocated fuel assembly, a dropped fuel assembly, and abnormally high SFP temperatures. Additionally, at the NRC staff's request, the licensee performed an analysis of mislocating a fuel assembly between the SFP wall and an adjacent rack. For each accident condition analyzed, the licensee used conservative assumptions and credited the SFP soluble boron concentration, as permitted by the NRC staff's guidance contained in an NRC letter dated August 19, 1998, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants.” For all the accident conditions analyzed, the licensee's calculations indicate that an SFP soluble boron concentration of 2475 ppm is adequate to ensure the maximum k_{eff} does not exceed 0.95

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[1] MkBI is a fuel type designation used by Duke

for the worst case SFP accident possible at McGuire, Units 1 and 2. The 2475 ppm concentration is credited to mitigate the consequences of a weir-gate drop accident in which fuel assemblies are crushed into an optimum criticality configuration. The licensee's TS requirements on SFP soluble boron concentration ensure that even for this highly unlikely event, the licensee will have sufficient subcriticality margin to ensure the SFP remains safe. The NRC staff reviewed the licensee's criticality analyses of normal unborated and borated cases as well as accident conditions and found that each meets the requirements of 10 CFR 50.68 and GDC 62.

In addition to its new criticality analyses of the spent fuel storage racks, the licensee proposed to change the licensing basis and storage requirements of the NFV. The licensee uses the NFV to store fresh fuel assemblies prior to loading them in the reactor for irradiation. The NFV contains racks which have a 21-inch nominal center-to-center spacing that effectively neutronically decouples one fuel assembly from adjacent assemblies. The licensee performed criticality analyses of the NFV using the following conservative assumptions: 1) all fuel designs that have been or are projected to be used in the McGuire reactors were evaluated; 2) all fuel assemblies were modeled as unirradiated and at the maximum permissible enrichment of 5 weight percent, except the Westinghouse Optimized Fuel Assembly (W-OFA) design; 3) the fuel assemblies are modeled without any location restrictions in the NFV; and 4) no credit was taken for spacer grid material, BPRAs, control rods, or other neutron poisons. For the W-OFA fuel assembly design, the licensee considered a maximum enrichment of 4.76 weight percent U-235 in the NFV criticality analyses as opposed to the 5 weight percent assumed for all other fresh fuel assembly types. The NRC staff finds this acceptable since the W-OFA design is a slightly older fuel assembly type that was used in cycles 4 through 9 at both McGuire Units. Therefore, all of the W-OFA fuel assemblies at McGuire, Units 1 and 2, have been irradiated and are stored in the SFP. The licensee no longer uses the W-OFA design and, therefore, will not be storing W-OFA fresh fuel assemblies in the NFV. The licensee performed criticality analyses of the various fresh fuel assembly types to be stored in the NFV using the computer codes described in Section 3.1.1 of this SE with the assumptions previously described. The licensee calculated that the NFV flooded with full-density unborated water would have a maximum 95/95 k_{eff} of 0.9498 and the NFV flooded with optimum-moderation unborated "water" would have a maximum 95/95 k_{eff} of 0.9618. The NRC staff reviewed the licensee's methodology and results of calculating the maximum k_{eff} of the NFV and found them acceptable because each satisfies the 10 CFR 50.68 and GDC 62 regulatory requirements as described in Section 2.1 of this SE.

3.2 Licensing Basis Change

The licensee, in its amendment request, proposed to change the McGuire, Units 1 and 2, SFPs licensing basis from a 10 CFR 70.24 exemption to 10 CFR 50.68 compliance. Therefore, the NRC staff reviewed the licensee's submittals to determine whether the SFPs complied with 10 CFR 50.68. Duke provided a detailed description of how it complied with each of the eight requirements in 10 CFR 50.68(b) in a letter dated July 29, 2004, in response to an RAI. These requirements include the following: 1) using plant procedures to ensure subcriticality and safe handling of fuel assemblies; 2) ensuring new fuel storage racks are subcritical by defined margins under both unborated and optimum moderation conditions; 3) verifying spent fuel storage racks are subcritical by defined margins under both borated and unborated conditions; 4) ensuring the quantity of Special Nuclear Material stored onsite is less than the quantity necessary for a critical mass; 5) providing radiation monitors in fuel storage and handling areas; 6) maintaining the maximum U-235 enrichment of fresh fuel assemblies less than or equal to 5 percent by weight; and 7) updating the Final Safety Analysis Report in a timely fashion after choosing to comply with 10 CFR 50.68.

The NRC staff reviewed each of the requirements that did not require a criticality analysis to verify that the licensee would meet the conditions. The NRC staff found that the licensee's responses as presented in a letter dated July 29, 2004, provided reasonable assurance that it would meet each of these requirements.

For requirements that required a criticality analysis to demonstrate compliance, the NRC staff reviewed the information provided by the licensee in support of this amendment request. In the NRC staff's review of this amendment request, it used the regulatory limits for k_{eff} that are described in 10 CFR 50.68 for both fresh and spent fuel storage racks. The NRC staff determined that the licensee's criticality analyses are both acceptable and in compliance with the regulatory limits. Therefore, the NRC staff finds that Duke will comply with all of the requirements of 10 CFR 50.68 and that a change in the licensing basis for the McGuire, Units 1 and 2, SFPs is appropriate and acceptable.

3.3 Credit for Boral® in Region 1 of the SFP

In its amendment request, the licensee stated that Region 1 of the SFP in both units was re-racked with new racks of equivalent design in accordance with the stipulations as set forth in 10 CFR 50.59. In addition, the new racks utilize Boral® as the neutron poison material.

By letter dated April 21, 2004, the NRC staff requested that the licensee discuss the procedures and tests to ensure that the Boral® panels in the new racks have the minimum Boron-10 (B-10) loading assumed in the criticality analyses. In its response dated June 9, 2004, the licensee stated that the new racks installed in 2003 have identical key physical parameters and dimensions except that the neutron poison material used in the new racks is Boral®. The neutron poison material in the previous racks had a minimum B-10 loading of 0.020 g/cm^2. The required minimum B-10 loading of the Boral® panel in the new racks is also 0.020 g/cm^2; however, the minimum purchase specification is 0.022 g/cm^2. This provides an additional margin of conservatism for the neutron poison material. In addition, the Boral® supplied by the manufacturer was examined and tested to certify that the specifications of the material complied with the requirements for materials, workmanship, boron content and markings.

The licensee further described the process for determining the boron loading of the Boral® material. This process was performed through density measurements and chemical analysis of samples from each lot of material. The number of samples was sufficient to verify that the B-10 areal density was greater than the guaranteed 0.022 g/cm^2 with a 95% probability and a 95 percent confidence level. The licensee indicated that all panels tested exceeded the certified minimum areal density of B-10. In addition, each Boral® piece was individually serialized and traceable to material and inspection documentation. The documentation for the Boral® material used in the racks included: chemical analysis for boron carbide, inspection data (minimum dimensions, visual appearance, and B-10 loading), wet chemistry results, certificate of compliance to purchase order requirements, results of density measurements and chemical analysis of samples cut from each lot, B_4C particle size distribution for each lot of B_4C

powder, spectroscopic (or similar) analysis of the boron in each lot of B_4C used, and composition of B_4C aluminum powder and aluminum plate used in each lot of panels.

Based on the licensee's response, the NRC staff finds that the manufacturing procedures and tests on the lots of Boral® material were sufficient to verify the minimum B-10 loading.
In addition to a visual inspection, the Boral® material was subjected to chemical tests to verify the minimum specifications for a neutron poison material credited in the criticality analysis. Therefore, the NRC staff concurs with the licensee assessment that the Boral® panels installed in the SFP racks have the minimum B-10 loading assumed in the criticality analysis.

Recent incidents of Boral® blistering have been reported by licensees utilizing this material in the SFP racks. At this time, the impact of the blistering on the expected performance of this material is not completely understood. Therefore, by letter dated April 21, 2004, the NRC staff requested the licensee to implement a TS Boral® coupon surveillance program to ensure consistent material performance assumed in its analysis. In addition, the NRC staff requested that the program monitor the physical and chemical properties in Boral® over time and include a description of the coupons used, the technique for measuring the initial B-10 content of the coupons, the surveillance frequency and justification, and the tests to be performed on the coupons. The NRC staff also requested that the program include provisions for informing the NRC staff, in writing, should the coupons reveal a change in material performance from that assumed in the analysis.

In its response dated June 9, 2004, the licensee indicated that, notwithstanding the stability and integrity of the Boral® demonstrated in more than 25 years of experience of wet storage applications, a surveillance program will be established at McGuire, Units 1 and 2, to monitor certain physical properties of the Boral®. The intent of this program is to detect the onset of any significant degradation in sufficient time to permit remedial action. As such, the licensee installed a coupon tree with ten Boral® coupons in the SFP of each unit and revised procedures to ensure that the coupons will be surrounded by freshly discharged fuel assemblies. In addition, a prearranged number of coupons will be removed on a prescribed schedule and tested to infer the stability and integrity of the Boral® in the SFP racks. The removed coupon may be tested for one or more of the following: visual examination and photographic documentation, dimensional measurement, weight and specific gravity, and neutron attenuation (B-10 areal density).

The licensee further responded that its Problem Identification Process (PIP) would be initiated in the event that an acceptance criterion for one or more of the above measurement parameters is not met. This process includes an evaluation of the impact on the criticality analysis, a determination if the condition meets NRC reporting requirements, and an engineering evaluation for further testing or necessary corrective actions. In addition, other programs/processes will be used, as warranted, to address and resolve the observed deviation.

To address the potential for blistering in the Boral®, the licensee responded that it tested representative coupons in a water environment at elevated temperatures. At least two coupons from each lot were immersed in demineralized water at 170EF for at least 45 days. The coupons showed no indications of swelling or delamination. The program will be defined within station procedures and directives consistent with other neutron poison material surveillance programs. The licensee further states that operating experience has shown that these surveillance programs have been effective in detecting the onset of any significant degradation with ample time to take corrective actions as necessary.

Based on its response, the NRC staff finds that the licensee's proposed Boral® coupon surveillance program has the appropriate elements for detecting significant degradation of the material. The program includes specific visual and chemical tests to support the assumptions of material performance in the criticality analysis. In addition, the NRC staff notes that the testing with neutron attenuation is appropriate for detecting changes in B-10 areal density. The NRC staff also notes that the implementation of the PIP and additional processes provide additional measures for addressing and evaluating the impact of deviations in expected performance of the material.

3.4 Ceasing of Commitment to Monitor Boraflex in Region 2 of the SFP

In an evaluation dated November 27, 2000, the NRC staff found the licensee's criticality analysis, which credited Boraflex in Region 2 of the SFP, acceptable provided the licensee complete the proposed Boraflex verification testing described in its UFSAR Section 16.9-9, "Spent Fuel Pool Storage Rack Poison Material." Since the current request does not credit the Boraflex material in the new criticality analysis, the licensee stated that it will no longer perform tests on the Boraflex material as described in selected licensee commitment 16.9.24, "Spent Fuel Pool Storage Rack Poison Material." The NRC staff finds that ceasing these tests on the Boraflex material acceptable based on the approval of the current criticality analysis.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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, and there has been no public comment on such finding (69 FR 55469). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The NRC staff reviewed the effects of the proposed McGuire, Units 1 and 2, SFP amendment using the appropriate requirements of 10 CFR 50.68 and GDC 62. The NRC staff found that the licensee's request provided reasonable assurance that under both normal and accident conditions the licensee would be able safely operate the plant and comply with the NRC regulations.

The NRC staff reviewed the licensee's request to credit Boral® in its criticality analysis for Region 1 of the SFP consistent with SRP 9.1.2. The NRC staff finds that the manufacturing procedures and tests ensure that the installed Boral® panels will have the minimum B-10 loading to support the analysis. Given the recent operating history of blister formation on the Boral® material used in SFP racks, the NRC staff requested and reviewed the licensee's proposed Boral® coupon surveillance program. The NRC staff finds that the proposed surveillance program has the appropriate elements for detecting significant degradation of the Boral® material.

In addition, the NRC staff finds that the licensee's additional processes will provide for evaluating the impact of deviations in the expected Boral® material performance. The NRC staff finds it acceptable to cease testing of the Boraflex material in Region 2 of the SFP as described in selected licensee commitment 16.9.24 because the licensee's current analysis does not credit the Boraflex material in SFP Region 2.

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.

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