June 15, 2006

Mr. Bruce H. Hamilton
Vice President, Oconee Site
Duke Power Company LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING CRITICALITY REQUIREMENTS FOR LOADING AND UNLOADING DRY SPENT FUEL STORAGE CANISTERS IN THE SPENT FUEL POOL (TAC NOS. MC0238, MD0239, AND MD0240)

Dear Mr. Hamilton:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 351, 353, and 352 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 1, 2006, supplemented April 26, 2006.

These amendments revise the TSs to reconcile the criticality requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, and 10 CFR Part 72 for loading and unloading spent fuel pool dry storage canisters in the spent fuel pool.

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/
Leonard N. Olshan, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 351 to DPR-38
2. Amendment No. 353 to DPR-47
3. Amendment No. 352 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

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DUKE POWER COMPANY LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 351
Renewed License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. DPR-38 filed by the Duke Power Company LLC (the licensee) dated March 1, 2006, supplemented April 26, 2006, 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 3.B of Renewed Facility Operating License No. DPR-38 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 351, are hereby incorporated in the 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L Raghavan for/
Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes
and License No. DPR-38 Changes

Date of Issuance: June 15, 2006

DUKE POWER COMPANY LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 353
Renewed License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. DPR-47 filed by the Duke Power Company LLC (the licensee) dated March 1, 2006, supplemented April 26, 2006, 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 3.B of Renewed Facility Operating License No. DPR-47 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 353, are hereby incorporated in the 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L Raghavan for/
Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes
and License No. DPR-47 Changes

Date of Issuance: June 15, 2006

DUKE POWER COMPANY LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 352
Renewed License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Renewed Facility Operating License No. DPR-55 filed by the Duke Power Company LLC(the licensee) dated March 1, 2006, supplemented April 26, 2006, complies with the standardsand 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 3.B of Renewed Facility Operating License No. DPR-55 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 352, are hereby incorporated in the 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L Raghavan for/
Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes
and License No. DPR-55 Changes

Date of Issuance: June 15, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 351

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 353

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 352

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License Pages	License Pages
License No. DPR-38, Pg 3	License No. DPR-38, Pg 3
License No. DPR-47, Pg 3	License No. DPR-47, Pg 3
License No. DPR-55, Pg 3	License No. DPR-55, Pg 3
TSs	TSs
iv	iv
3.7.12-1	3.7.12-1
--	3.7.18-1
--	3.7.18-2
--	4.0-3
iv (Bases)	iv
B 3.7.12-1	B 3.7.12-1
B 3.7.12-2	B 3.7.12-2
B 3.7.12-3	B 3.7.12-3
B 3.7.12-4	B 3.7.12-4
B 3.7.12-5	--
--	B 3.7.18-1
--	B 3.7.18-2
--	B 3.7.18-3
--	B 3.7.18-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 351 TO RENEWED FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 353 TO RENEWED FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 352 TO RENEWED FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated March 1, 2006 (Ref. 1), and supplemented by letter dated April 26, 2006 (Ref. 2), Duke Power Company LLC (Duke, the licensee) requested amendments to the Technical Specifications (TS) for Oconee Nuclear Station, Units 1, 2, and 3 (Oconee), with respect to loading and unloading spent fuel dry storage canisters (DSC) in the spent fuel pool (SFP).

The supplement dated April 26, 2006, provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

In March 2005, the Nuclear Regulatory Commission (NRC) issued Regulatory Issue Summary (RIS) 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," (Ref. 3), to (1) alert addressees to findings at pressurized-water reactor (PWR) facilities suggesting that the SFP licensing and design bases and applicable regulatory requirements may not be met during loading, unloading, and handling of DSCs in the SFPs; (2) emphasize the importance of maintaining subcritical conditions for spent fuel storage in a moderated environment; and (3) encourage addressees to review the current SFP and independent spent fuel storage installation (ISFSI) licensing and design bases at their facilities to ensure compliance of regulations during DSC loading, unloading, and handling operations. In response to RIS 2005-05, the licensee determined a license amendment was required to reconcile the requirements of Title 10 of the *Code Federal Regulations* (10 CFR), Part 50, Section 50.68, and 10 CFR Part 72.

2.0 REGULATORY EVALUATION

While in the SFP for wet loading operation, both 10 CFR 50.68 and 10 CFR Part 72 requirements pertaining to criticality control apply to DSC loading operations. General Design Criterion (GDC) 62 in Appendix A to 10 CFR Part 50 specifies that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The acceptance criteria specified in 10 CFR Section 50.68(b)(1) and (4) for criticality prevention in the SFP that are applicable to the licensee's proposed amendment are, respectively:

> (1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water; and
>
> (4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

Under 10 CFR 72.124, "Criteria for nuclear criticality safety," the NRC regulates DSC storage activities to ensure that subcriticality is maintained during the handling, packaging, transfer, and storage of spent fuel assemblies. The NRC regulations for DSC criticality prevention rely on favorable geometric configurations and fixed neutron absorbers. However, unlike 10 CFR 50.68, the 10 CFR Part 72 regulations for criticality prevention in DSCs allow licensees to credit the SFP soluble boron for maintaining subcritical conditions during DSC loading, unloading, and handling operations in the SFP. Therefore, many DSC designs have incorporated soluble boron credit in lieu of a burnup credit as a means of increasing DSC storage capacity while maintaining subcritical conditions. Duke's license amendment request (LAR) proposes to demonstrate that it can satisfy the applicable 10 CFR 50.68 criticality prevention requirements, with a burnup credit, during DSC loading, unloading, and handling operations in the SFP.

The NRC defines acceptable methodologies for performing criticality analyses in the following documents:

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

2. 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," August 19, 1998, (Ref. 5).

The NRC staff used these documents to assist in its review of the licensee's LAR to ensure compliance with GDC 62 and 10 CFR 50.68.

3.0 TECHNICAL EVALUATION

3.1 Spent Fuel Cask System Description

Oconee uses the NUHOMS® dry spent fuel storage system at its ISFSI. Forty NUHOMS®-24P DSCs have been loaded under Duke's specific license (SNM-2503). Another forty-four NUHOMS®-24P DSCs have been loaded under Duke's general license. Certificate of Compliance (CoC) 72 -1004 is applicable to the DSCs loaded under the general license. For future loadings, Oconee will use the NUHOMS®-24PHB, which has been approved by the NRC as Amendment 6 to CoC 72 -1004 (ML040128014). The NUHOMS®-24P and NUHOMS®-24PHB DSCs will hold 24 fuel assemblies. Neither contain any fixed neutron absorber. Figure 1 of Enclosure 3 to Reference 1 shows the configuration of the fuel assembly loading into the NUHOMS®-24P and NUHOMS®-24PHB DSCs.

3.2 Description of Proposed TS Changes

Attachment 1 to the March 1, 2006, (Ref. 1) letter provided a mark-up of the TSs and corresponding bases pages. The following is the descriptive list of the changes proposed by the LAR:

1. Modification of the Applicability of TS 3.7.12, "Spent Fuel Pool Boron Concentration"
   The modification makes the TS applicable during DSC loading activity. The acceptability of the modified TS Limited Condition for Operation 3.7.12 is demonstrated by the licensee's criticality analysis for compliance with the subcriticality criteria of GDC 62 and 10 CFR 50.68. The NRC staff's evaluation of the criticality analysis is described in Section 3.3 below.

2. Addition of TS 3.7.18, "Dry Spent Fuel Storage Cask Loading and Unloading"
   The licensee proposed to add TS 3.7.18, "(Dry Spent Fuel Storage Cask Loading and Unloading," specifying burnup versus enrichment curve (TS Table 3.7.18-1) for fuel assemblies located in the dry spent fuel storage DSC in the SFP. TS 3.7.18 also includes required actions when these LCOs are not met.

   The acceptability of the new TS 3.7.18 is demonstrated by the licensee's criticality analysis for compliance with the subcriticality criteria of GDC 62 and 10 CFR 50.68. The NRC staff evaluation of the criticality analysis is described in Section 3.3 below.

3. Addition of TS 4.4, "Dry Spent Fuel Storage Cask Loading and Unloading"
   The licensee proposed to add TS 4.4, "(Dry Spent Fuel Storage Cask Loading and Unloading," to specify that the spent fuel casks are designed and shall be maintained with:

   a. Fuel assemblies having a maximum (*nominal*) U-235 enrichment of 5.0 weight percent (w/o);

   b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the (Updated Final Safety Analysis Report) UFSAR;

c. k_{eff} * 0.95 if fully flooded with borated water * 430 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR. Maintaining the normal spent fuel pool boron concentration within the TS limits assures k_{eff} * 0.95 for any accident condition;

d. Dry spent fuel storage cask designs limited to NUHOMS®-24P and NUHOMS®-24PHB.

Added Section 4.4 describes characteristics of the spent fuel DSC loading in the SFP. Items b and c specify the commitment to meet 10 CFR 50.68 reactivity limits. Items a and d specify the requirements regarding enrichment limit, and spacing of fuel assemblies loaded in the spent fuel DSC to assure compliance with the subcriticality requirements. Therefore, added TS 4.4 is acceptable.

3.3 Criticality Analysis

3.3.1 Criticality Analysis Codes

The main neutronics codes employed in the licensee's NUHOMS®-24P/24PHB DSC criticality analysis are SCALE 4.4/KENO V.a and CASMO-3/SIMULATE-3. KENO V.a is a 3-D Monte Carlo criticality module in the SCALE 4.4 package. The licensee used SCALE 4.4/KENO V.a for an explicit 3-D evaluation of the NUHOMS®-24P/24PHB DSC loaded with unirradiated fuel assemblies. The SCALE 4.4/KENO V.a computations are performed to confirm the conservatism of a simplified uniform-array DSC model.

CASMO-3 is a 2-D transport code that performs fuel criticality and depletion calculations, using a 70-group cross-section library that is based on ENDF/B-IV. CASMO-3 also produces nodal macro-group cross-sections that can be used by SIMULATE-3, its counterpart 3-D nodal diffusion code, for applications involving arrays of fuel assemblies with varying enrichments or burnups. CASMO-3/SIMULATE-3 is used for all DSC irradiated-fuel cases.
These codes are well-suited to wet fuel storage criticality applications and have been extensively benchmarked to critical experiments and reactor operational data. Their use by Duke for the Oconee dry spent fuel storage DSC loading and unloading LAR is consistent with how Duke employed them for the Oconee SFP amendment (Ref. 6), and the more recent McGuire SFP amendment (Ref. 7).

3.3.2 Bias and Uncertainty

The NRC SFP criticality analysis guidance (Ref. 5) specifies that the maximum k_{eff} value for the criticality analysis should be the summation of the calculated nominal k_{eff}, the bias in criticality analysis methods, manufacturing and calculational uncertainties, and the correction for the effect of the axial distribution in burnup when credit for burnup is taken. It further provides that uncertainties should be determined for proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should indicate all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be statistically combined provided they are independent variations. Combinations of the two methods may also be used. The licensee's modeling approach for the NUHOMS®-24P/24PHB DSC criticality analysis is a combination of the two methods.

SCALE 4.4/KENO V.a Bias and Uncertainty

As described in the letter dated March 1, 2006 (Ref. 1), and supplemented by letter dated April 26, 2006, (Ref. 2), Duke determined SCALE 4.4/KENO V.a calculational biases and uncertainties for both the 44-group and 238-group cross-section libraries included with the SCALE 4.4 package. The licensee's biases and uncertainties were determined by benchmarking against 58 critical experiments carried out by Pacific Northwest Laboratories. These are the same methods and experiments the licensee used to determine the bias and uncertainties for the Oconee SFP criticality analysis (Ref. 6). The resultant bias and uncertainty are slightly different, which is attributed to updated computer codes. By comparison, these are also the same method and experiments the licensee used to determine the bias and uncertainties for the McGuire SFP criticality analysis (Ref. 7). The resultant bias and uncertainty are identical. The licensee determined the SCALE 4.4/KENO V.a calculational bias to be +0.0064 *k and the uncertainty to be ±0.0066 *k.

CASMO-3/SIMULATE-3 Bias and Uncertainty

As described in the letter dated March 1, 2006 (Ref. 1), and supplemented by letter dated April 26, 2006 (Ref. 2), Duke determined a CASMO-3/SIMULATE-3 calculational bias and uncertainty for the 70-group cross-section library included with the CASMO-3/SIMULATE-3 package. The licensee's bias and uncertainty were determined by benchmarking CASMO-3/SIMULATE-3 against 10 Babcock & Wilcox (B&W) critical experiments. These are the same methods and experiments the licensee used to determine the bias and uncertainties for the Oconee SFP criticality analysis (Ref. 6). The resultant bias and uncertainty are slightly different, which is attributed to updated computer codes. Similarly, these are the same methods and experiments the licensee used to determine the bias and uncertainties for the McGuire SFP criticality analysis (Ref. 7). The resultant bias and uncertainty are identical. The licensee determined the CASMO-3/SIMULATE-3 calculational bias to be -0.00142 *k and the uncertainty to be ±0.0121 *k.

Spent Fuel Burnup Uncertainties

In the burnup credit analyses, biases are added to the bounding criticality safety predictions to account for the uncertainties associated with (1) the axial burnup distribution in the active fuel length, and (2) the assembly burnup accuracy. The dominant bias is from the axial burnup effects.

The burnup credit calculations were performed assuming a uniform burnup profile throughout the active fuel length that results in the over-prediction of burnup at the ends of the fuel assembly and under-prediction of burnup in the fuel mid-region. The "axial end-effect" bias, i.e., the difference of the k_{eff} values between the axial burnup profile and the uniform burnup assumption, needs to be applied to the burnup credit calculations to account for the increase in reactivity. Generic analyses confirmed minor and generally negative reactivity effects of the axially distributed burnup at values less than 20,000 MWD/MTU. As a result, CASMO-3/SIMULATE-3 calculations with less than 20,000 MWD/MTU do not contain an axial bias.

The highest burnup evaluated in the effort was 43.77 GWD/MTU, to which the axial bias uncertainty of +0.0245 *k is applied.

Another major contributor to the bounding uncertainty is the bias in the assembly burnup to account for the uncertainty in the fuel depletion calculations. The NRC guidance document (Ref. 5) stated an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption. The licensee applied this option and then applied the largest uncertainty to all cases.

Fuel Mechanical Uncertainty

To determine the fuel mechanical uncertainty, the licensee performed an analysis that evaluated all of the fuel designs that are currently licensed to be loaded into its NUHOMS$_{®}$-24P/24PHB DSCs to determine a bounding set of mechanical parameters. The tolerances for those parameters were then used to calculate an associated reactivity uncertainty. Those uncertainties were then statistically combined into the fuel mechanical uncertainty. Two fuel mechanical uncertainties were determined, one with the moderator unborated and one with the moderator borated to 430 ppm.

DSC Mechanical Uncertainty

To determine the DSC mechanical uncertainty, the licensee performed an analysis that identified the applicable mechanical parameters associated with the NUHOMS$_{®}$-24P/24PHB DSCs. The mechanical tolerances for those parameters were then used to calculate an associated reactivity uncertainty for each parameter. Those uncertainties were then statistically combined into the DSC mechanical uncertainty. Two DSC mechanical uncertainties were determined, one with the moderator unborated and one with the moderator borated to 430 ppm.

3.3.3 DSC Loading Criticality Analysis

To simplify the analysis, the licensee performed all criticality analyses using CASMO-3/SIMULATE-3. In order to use the 2-D CASMO-3/SIMULATE-3 computer code, the licensee performed a comparison analysis with the 3-D SCALE 4.4/KENO V.a computer code. The licensee performed SCALE 4.4/KENO V.a computer code cases for unirradiated fuel assemblies at both high and low enrichments and moderator temperatures while explicitly modeling the 3-D NUHOMS$_{®}$-24P/24PHB DSC. The licensee also performed SCALE 4.4/KENO V.a computer code cases for unirradiated fuel assemblies at both high and low enrichments and moderator temperatures while modeling the simplified 2-D NUHOMS$_{®}$-24P/24PHB DSC model used in the subsequent CASMO-3/SIMULATE-3 computer code cases. The high and low enrichments and moderator temperatures used in the analyses are sufficient to encompass expected conditions in the SFP. The analyses showed that in each instance the 2-D CASMO-3/SIMULATE-3 computer code predicted the highest keff, thereby providing reasonable assurance that the use of the 2-D CASMO-3/SIMULATE-3 computer code to perform the NUHOMS$_{®}$-24P/24PHB DSC SFP loading and unloading criticality analyses at Oconee would provide appropriately conservative results.

To demonstrate compliance with the reactivity criteria of GDC 62 and 10 CFR 50.68, the licensee performed the criticality analyses of the NUHOMS$_{®}$-24P/24PHB DSC loading and unloading in the SFP for both normal and accident conditions. Consistent with the methodology of using bounding parameters, the criticality analyses were performed with the conservative assumptions that tend to maximize the reactivity. These assumptions include:

- Ignoring the negative CASMO-3/SIMULATE-3 calculational bias.
- Not taking credit for any burnable poison rod assemblies (BPRA) that might be loaded into the DSC. This assumption is conservative with respect to any negative reactivity impact the remaining burnable poison would have. However, the NUHOMS®-24P/24PHB DSC is over-moderated. The NRC staff asked the licensee if loading the NUHOMS®-24P/24PHB DSC with BPRAs would increase the overall reactivity due to the moderator they would displace. The licensee was able to show there was a net decrease in reactivity with the NUHOMS®-24P/24PHB DSC loaded with the maximum number of BPRAs possible, even if the BRPAs were fully depleted not taking credit for the negative reactivity effect of the fuel assembly grid spacers.
- The use of conservative parameters when determining fuel assembly burnup.
- Full DSC reflection in the radial direction.

To demonstrate compliance with the subcriticality criteria of 10 CFR 50.68 that k_{eff} < 1.0 if flooded with unborated water, the licensee set a target total 95/95 k_{eff} of 0.9980. The licensee then subtracted the various biases and uncertainties from the targeted total 95/95 k_{eff} to establish a nominal k_{eff}. The nominal k_{eff} was then used to determine a corresponding enrichment/burnup combination that would result in the nominal k_{eff}. The results are shown in Table 6 and Figure 4 of Reference 1. By this method, the licensee was able to demonstrate that by following these limits the DSC k_{eff} will remain < 1.0 if flooded with unborated water.

To demonstrate compliance with the subcriticality criteria of 10 CFR 50.68 that k_{eff} be less than or equal to 0.95 if flooded with borated water, the licensee performed analyses with the same enrichment/burnup combinations as above, but with the DSC borated to 430 ppm. In this analysis the licensee substituted the fuel and DSC mechanical uncertainties for the moderator borated to 430 ppm. The other biases and uncertainties remained the same. The licensee concluded that the 5.0 wt % U-235 with 43.77 GWD/MTU of burnup and 430 ppm of soluble boron, will have the highest Total k_{eff} at 0.9264. In paragraph 3 of Section 6.5 of Reference 1, the licensee makes the statement, "Using the fuel assembly burnup requirements shown in Table 6, DSC calculations with 430 ppm boron yield the highest nominal k_{eff} (0.8645) at 5.0 wt % U-235 (as expected per the Reference 15 report)." However, NUREG/CR-6683 (Reference 15 of the licensee's LAR) would indicate that calculations with 430 ppm boron yield the highest total, rather than nominal, keff at 5.0 wt % U-235 with 43.77 GWD/MTU of burnup. Since the total keff is the value of interest, the licensee's overall conclusion remains valid, and since that total keff is less than 0.95, the results are acceptable.

3.3.4 Accident Analysis

In Reference 1 the licensee identified that the most limiting accident was the misloading of an unirradiated 5.0 w/o U-235 fuel assembly into the DSC. In response to a question from the NRC staff, the licensee provided additional details in Reference 2 to support the NRC staff 's review. In performing the accident analysis the licensee utilized the double contingency principle from Reference 5, wherein a licensee may credit the amount of soluble boron normally in the SFP (for Oconee, this value is 2200 ppm).

For the misloading event described above the licensee evaluated all fuel types currently in use at Oconee, including those not currently licensed for storage in the NUHOMS®-24P/24PHB DSC at Oconee. The licensee determined the maximum boron concentration required to maintain k_{eff} * 0.95 was 630 ppm.

In Reference 2 the licensee described the controls in place to ensure that the misloading of only one fuel assembly need be considered. In Reference 2 the licensee provided a discussion of other potential accident scenarios and a justification for why they are bounded by the misloading event.

The NRC staff evaluated the licensee’s criticality analyses methodology, and concluded that the use of these bounding parameters to produce the Maximum k_{eff} is consistent with the NRC criticality analyses guidance, and is acceptable.

The NRC staff reviewed the licensee’s LAR to reconcile the criticality control requirements of 10 CFR 50.68 and 10 CFR 72.124. That reconciliation has led to the revision of existing TSs to make the SFP soluble boron concentration applicable during DSC loading and unloading, the addition of new TSs to prescribe a burnup verses enrichment curve for fuel assemblies currently licensed to be stored in the NUHOMS®-24P/24PHB DSC, and the addition of TSs to describe the characteristics of the DSC loading in the SFP. Based on its review of the licensee’s criticality analyses, provided in References 1 and 2, the NRC staff concludes that the proposed changes for the spent fuel cask loading operations in the SFP meet appropriate subcriticality requirements of 10 CFR 50.68 and GDC 62. The NRC staff found that the licensee’s amendment request provides reasonable assurance that under both normal and accident/upset conditions, the licensee would be able to safely operate the plant and comply with NRC regulations. Therefore, the staff finds the LAR acceptable.

4.0 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.

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 and change surveillance requirements. 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 (71 FR 18373). 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 has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the public health and safety 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) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from Bruce Hamilton, Duke Power Company LLC, (Duke), USNRC, "Oconee Nuclear Site, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, License Amendment Request to Reconcile 10 CFR 50 and 10 CFR 72 Criticality Requirements for Loading and Unloading Dry Spent Fuel Storage Canisters in the Spent Fuel Pool, License Amendment Request No. 2005-009," dated March 1, 2006 (ADAMS ML060720030).

2. Letter from Bruce Hamilton, Duke, USNRC, "Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC, Oconee Nuclear Site, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, License Amendment Request to Reconcile 10 CFR 50 and 10 CFR 72 Criticality Requirements for Loading and Unloading Dry Spent Fuel Storage Canisters in the Spent Fuel Pool - Duke Response to NRC Request for Additional Information, License Amendment Request (LAR) No. 2005-009," dated April 26, 2006 (ADAMS ML061240463 & ML061240464).

3. NRC Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," March 23, 2005 (ADAMS ML043500532).

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

5. 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," August 19, 1998 (ADAMS ML003728001).

6. Letter from L. Olshan, USNRC, to W. McCollum, Duke, "Oconee Nuclear Station Units 1, 2, and 3 Re: Issuance of Amendments (TAC Nos. MB0894, MB0895, and MB0896)," April 22, 2002 (ADAMS ML020930470).

7. Letter from J. Shea, USNRC, to G. Peterson, Duke, "McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MC0945 and MC0946)," March 17, 2005 (ADAMS ML050280409).

Principal Contributor: K. Wood

Date: June 15, 2006

Oconee Nuclear Station, Units 1, 2, and 3

cc:

Ms. Lisa F. Vaughn
Duke Power Company LLC
526 South Church Street
P. O. Box 1006
Mail Code EC07H
Charlotte, North Carolina 28201-1006

Manager, LIS
NUS Corporation
2650 McCormick Dr., 3rd Floor
Clearwater, FL 34619-1035

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
7812B Rochester Highway
Seneca, SC 29672

Mr. Henry Porter, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Dept. of Health and Env. Control
2600 Bull St.
Columbia, SC 29201-1708

Mr. Michael A. Schoppman
Framatome ANP
1911 North Ft. Myer Dr.
Suite 705
Rosslyn, VA 22209

Mr. B. G. Davenport
Regulatory Compliance Manager
Oconee Nuclear Site
Duke Energy Corporation
ON03RC
7800 Rochester Highway
Seneca, SC 29672

Ms. Karen E. Long
Assistant Attorney General
NC Department of Justice
P.O. Box 629
Raleigh, NC 27602

Mr. R. L. Gill, Jr.
Manager - Nuclear Regulatory
Issues and Industry Affairs
Duke Power Company LLC
526 S. Church St.
Mail Stop EC05P
Charlotte, NC 28202

Division of Radiation Protection
NC Dept of Environment, Health, & Natural
Resources
3825 Barrett Dr.
Raleigh, NC 27609-7721

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, NC 28210

Mr. Henry Barron
Group Vice President, Nuclear Generation
and Chief Nuclear Officer
P.O. Box 1006-EC07H
Charlotte, NC 28201-1006